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Status of High Temperature Reactor Development in the Federal Republic of Germany

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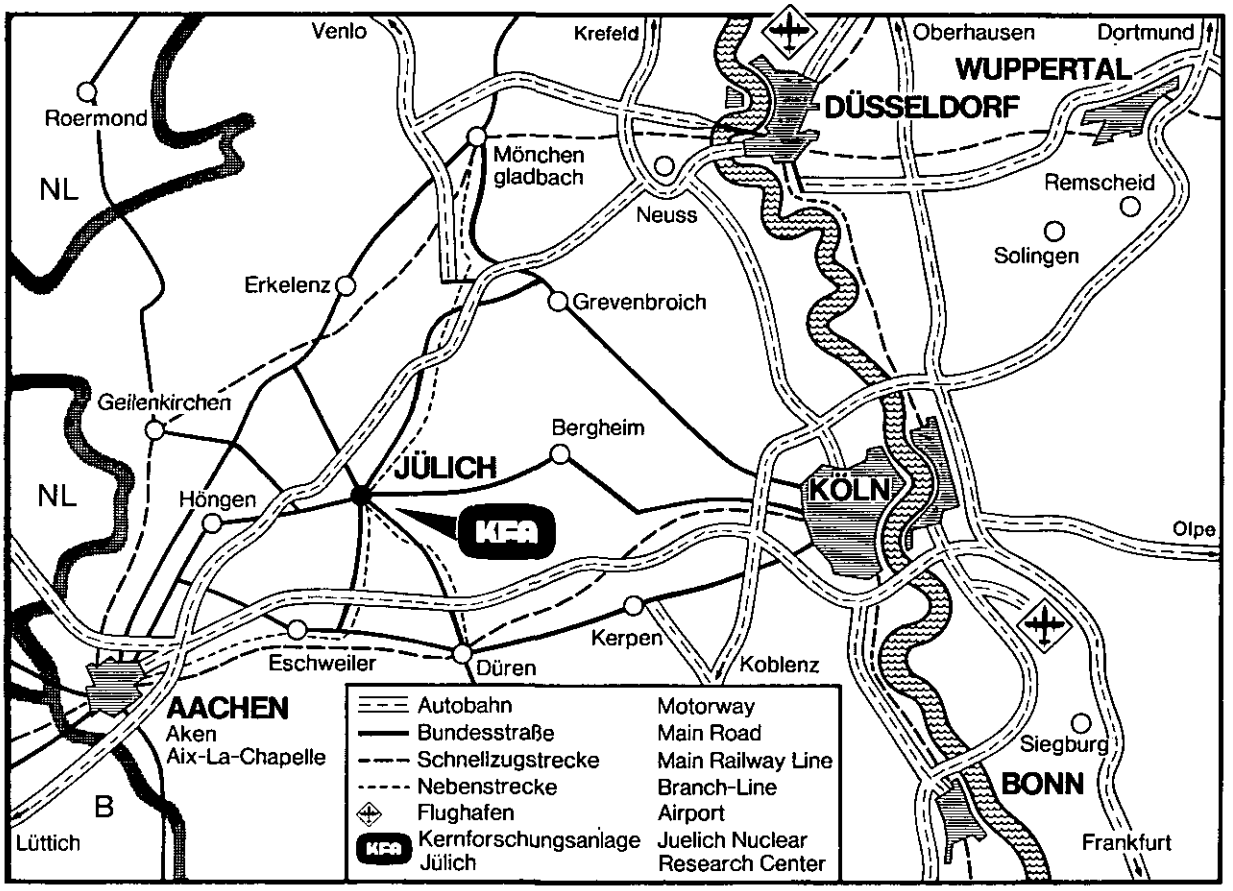

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Status of High Temperature Reactor Development in the Federal Republic of Germany

Der Bericht wurde zusammengestellt von

P. Engelmann

Foreword

The Swedish AB Atomenergi held a HTR Information Seminar in Stockholm on January 11, 1978. At this seminar the status of the High Temperature Reactor development program in the Federal Republic of Germany was presented in a series of papers. This report containing the papers in english language makes the information also available to other parties outside Germany who are interested in the HTR development. The papers deal with the German HTR-program, the experience from construction and operation of HTR-plants, the HTR-technology, especially the nuclear system and its application for power production and process heat, the nuclear coal gasification and nuclear long distance energy transport, the HTR fuel cycles, the safety of HTR plants and nuclear process heat plants, an economic evaluation of the HTR as a power plant and/or source of process heat and the introduction strategy of this reactor system in the FRG. The review represents the status at the beginning of 1978.

Vorwort

Auf Einladung der schwedischen AB Atomenergi fand am 11. Januar 1978 in Stockholm ein HTR-Informationsseminar statt. Hierbei wurde eine Statusübersicht über das deutsche Hochtemperaturreaktor-Entwicklungsprogramm gegeben. Die Zusammenstellung der Vorträge, die in englischer Sprache gehalten wurden, als Bericht soll diese Information einem größeren Interessentenkreis, vor allen Dingen im Ausland, zugänglich machen. Die Vorträge informieren im Einzelnen über das deutsche HTR-Programm, die Bau- und Betriebserfahrung mit HTR-Anlagen, die HTR-Technologie in ihrer Anwendung für Kernkraftwerke und Prozeßwärmeanlagen, die Nukleare Kohlevergasung und die Nukleare Fernenergie, den HTR-Brennstoffzyklus, die Sicherheit von Hochtemperaturreaktoren und nuklearen Prozeßwärmeanlagen, Fragen der Wirtschaftlichkeit von HTR-Kraftwerken und Prozeßwärmeanlagen sowie die Markteinführungsstrategie für HTR-Anlagen in Deutschland. Die Statusübersicht bezieht sich auf den Stand Anfang des Jahres 1978.

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THE GERMAN HTR PROGRAM

by P. Engelmann, KFA Jülich

For almost 2 decades the helium cooled HTR is being developed in the FRG in addition to the LMFBR as one of the two advanced reactor systems which both have specific advantages as compared to the LWR. While the Fast Breeder Reactor is well known all over the world and today a subject of heavy debate in a number of countries, the HTR is a reactor system which is promoted only by some industrialized countries.

I therefore will start my survey on the German HTR program with a brief review of the main characteristics of high temperature reactors (table 1).

The HTR may be considered as the nuclear system which shows the highest degree of environmental compatibility. This is due to the following facts:

- under normal operating condition the amount of radioactivity released is very low due to the high fission product retention of the coated particle fuel and the ceramic fuel elements
- under abnormal conditions the system's inherent safety features prevent a catastrophic failure endangering the general public even in so-called hypothetical accidents
- because of the high operating temperatures the heat losses in electricity production are relatively small reducing the waste heat problem. Dry air cooling towers can be used without an undue economic penalty.

Due to the helium outlet temperatures of 750 to 1000^{o}C HTRs have a wide application potential. On the one hand there is the electricity generation using either a conventional steam cycle or a direct helium cycle with a closed gas turbine. Net efficiencies of 39 % to 44 % may be obtained. Especially the direct cycle HTR (HHT) in combination with a dry air cooling tower offers the possibility of providing hot water at temperatures around 100^{o}C for district heating. without reduction of electric power output.

The upper temperature range, however, can best be used to supply energy for chemical processes in the production of secondary fuels such as substitute natural gas, synthesis gas, hydrogen, and methanol. Of special interest in Germany is the gasification of hard coal and lignite using nuclear process heat. In Sweden the gasification of peat might be more interesting. In Japan, the application of high temperature nuclear process heat is primarily seen in the steel industry. But there will be many more uses of NPH, once the HTR becomes commercially available. In most cases, the NPH offers three advantages as compared to fossil heat: it reduces the consumption of fossil fuels, it reduces the environmetal burden otherwise encountered by burning coal or oil, and it will eventually be more economic.

A further advantage of HTRs I would like to mention here is the high nuclear fuel conversion efficiency. Especially in the U 233- Th 232 fuel cycle high conversion ratios can be obtained expanding by factors 5-10 as compared to a present day LWR the energy which can be extracted

from a given amount of uranium. The flexibility in the mode of fuelling the reactor allows an adaptation to various boundary conditions such as availability or non-availability of highly enriched uranium, existence or non-existence of a closed fuel cycle etc.

Several papers during this information seminar will describe the HTR features in more detail and will explain the construction principles leading to these characteristics.

I will now turn to the HTR work in Germany giving a brief outline of the history and the present status.

Following a pre-phase of basic r+d and conceptual design work the first experimental HTR power plant AVR was built at Jülich. Construction started in 1960, operation in 1967. This 15 MWe plant has demonstrated - in its 10 years of operation - the feasibility of the pebble bed HTR concept, the excellent safety features and the reliability of its components. The helium outlet temperature was raised in 1972 from its original value of 750^{o}C to 850^{o}C and in 1974 to 950^{o}C. Even at this extremely high temperature the radioactivity level in the primary circuit remained very low. In 1976 a time availability of 92 % was reached, the average availability over the first 10 years of operation was 78 %. The fuel elements have reached by now burn ups of up to 190.000 MWd/t. A great variety of fuel concepts could be tested in the AVR on a statistical basis, as the number of fuel elements in the reactor is about 90.000.

Based on the AVR experience a 300 MWe prototype station, THTR 300, was designed in the late 60es. Construction of this plant started near Hamm in 1972. The plant is being built by the consortium HRB, BBC, Nukem for a group of utilities, HKG, headed by VEW of Dortmund. Completion of the plant was originally expected in 1977/78 but major delayes were encountered due to changes in licensing requirements during the construction period so that the exact time of completion can not be defined at this moment.

In the early 70es the US company Gulf General Atomic started commercial marketing of a similar HTR system, the HTGR with prismatic fuel elements. HRB, a 45 % daughter of GAC, at that time saw the chance of marketing the HTGR system also in Germany at lower cost and risk than its own design. Unfortunately the GAC approach was not successful.

Therefore in 1976 work was reactivated in Germany on the concept of a commercial size steam cycle HTR with spherical fuel elements (HTR-K). In June 1977 this concept was compared with that of a direct cycle HTR plant (HHT) of the same thermal capacity of 3000 MW. The direct cycle concept work was started in 1972 as an advanced HTR concept based on HTGR technology in a close cooperation of German and Swiss partners on the one hand and GAC on the other.

A comprehensive r+d program for this HHT system is still under way, including the construction of a very large Helium test facility, HHV, at KFA, for full size testing of major components at 850°C.

The comparison of the HTR-K and HHT showed, that the steam cycle HTR is the technically easier approach, the direct cycle HHT, however, has the larger potential. HRB/BBC therefore proposed to build a 1500 MW_{th} HHT demonstration reactor as the next HTR power station in Germany. Design work in the second half of 1977 was oriented to this HHT demonstration station with a pebble bed core.

The FRG - as other industrialized countries - is confronted with an increasing dependence on imported primary energy, especially oil and natural gas, putting a burden on the balance of payment and leading to risks in the continuity of supply. In the FRG in the medium term the gasification of lignite and hard coal, using nuclear heat, is seen as a viable way to cover an increasing part of the energy demand.

In cooperation of coal industry (Bergbau-Forschung GmbH, BF, and Rheinische Braunkohlenwerke AG, RBW), reactor manufactures (Gesellschaft für Hochtemperatur-Technik mbH, GHT, and Hochtemperatur-Reaktorbau GmbH, HRB) and the Kernforschungsanlage Jülich GmbH (KFA) a Project Prototype Nuclear Process-Heat (PNP) was formally established in 1975, aiming at the development and construction of a demonstration plant for nuclear coal gasification using a HTR as heat source.

In the first phase of the project, which was completed by the end of 1976, the concepts and design parameters of a commercial size coal gasification plant were fixed and the design of a smaller demonstration plant was initiated. The reactor design is based on the pebble bed concept with OTTO loading: the spherical fuel elements enter the reactor at the top and then flow once through the reactor. In this way a uniform burn-up of the fuel elements is achieved and neutron flux-, power-, and temperature distributions are such that a helium outlet temperature of 950°C can be obtained with a maximum temperature of the coated particles of only 1050°C - 1100°C.

In support of the reactor projects, comprehensive r+d programs are being carried out both at KFA and industry. At Jülich, the large He-test facility HHV with a throughput of 200 kg He/s at 850°C is now nearing completion. Another important area is the high temperature materials testing, where new facilities have come into operation and additional one's are being installed for long time testing of alloys under He and process gas conditions at temperatures up to 950°C.

Last year all our efforts in the area of fuel and graphite development and on reprocessing were concentrated in a new project HBK under the lead of KFA with NUKEM, HOBEG, SIGRI, Gelsenberg AG as well as HRB and GHT as partners. We hope that the HTR fuel cycle can be closed on the scale of a pilot plant serving about 5000 MWe reactor capacity by the end of the century and that this pilot plant can be built as an international venture.

Let me finish my introduction and survey by giving you some figures on our man-power capacity and budget. The man-power at KFA and industry presently engaged in HTR activities by far exceeds 1000 people. The federal government and the state of NRW have spent so far 1.800 Mio DM for the HTR development, including 170 Mio DM for the construction and operation of AVR, 800 Mio DM for the construction of THTR, and 300 Mio DM for fuel and material development. In 1976, about 250 Mio DM were spent, the budget for 1978 is 280 Mio DM.

It is expected, that public money at the same level will be available in the future. The FRG, therefore, can carry out a major development program in the HTR field. Nevertheless it is considered essential to broaden the basis and to reduce the risk of failure by international cooperation. In February 1977 an umbrella agreement for the cooperation in the field of gas cooled reactors was concluded between the USERDA and the German BMFT. Later in 1977 France and Switzerland became members of this governmental agreement, the implementation of which will also ease r+d- and industrial arrangements between companies of these four countries. Bilateral agreements also exist and/or are in preparation between German institutions and those in Austria and Japan.

Dr. Balthesen this afternoon will give an outline of our ideas on the market introduction of the HTR for both power production and process heat.

CHARACTERISTICS OF HTR's:

- good environmental compatibility due to
  - low emission of radioactivity
  - low risk because of inherent safety
  - low amount of waste heat
- wide application potential due to high coolant temperatures
  - high efficiency electricity generation
  - combination of power and district heating
    (in case of HHT and dry air cooling without loss in power production)
  - production of various forms of secondary energy
    (SNG, H_2, methanol, synthesis gas)
  - iron ore reduction
- high nuclear fuel conversion efficiency
  - expanding energy output of uranium
  - use of thorium
  - flexibility in the mode of fuelling

Hochtemperatur-Reaktorbau GmbH

Construction and Operating Experience with High Temperature Reactors in the Federal Republic of Germany

H.-W. Müller
H. Vollmer

Paper submitted to the HTR Seminar in Stockholm,
January 11, 1978

1. Introduction

In the Federal Republic of Germany the essential experience with High Temperature Reactors (HTR) has been gained with the nuclear power plants AVR and THTR.

The AVR (Fig. 1) is an experimental reactor with 15 MW electrical output, constructed by Brown, Boveri/Krupp Reaktorbau GmbH (BBK)* next to the site of the Kernforschungsanlage, Jülich (KFA), and operated by the Arbeitsgemeinschaft Versuchsreaktor (AVR) GmbH.

The AVR was conceived as a large-scale experiment in which the following aims were pursued:

- Demonstration of the feasibility of HTR's with a pebble bed core
- functional testing of main components, in particular the fuel element, under HTR-specific conditions
- demonstration of the safe and reliable operation of an HTR as a power plant.

The construction of the AVR was started in August 1961, on December 17, 1967 the AVR supplied first electricity into the public grid. The experimental reactor has since demonstrated an extremely successful operation over lo years, as is indicated in more detail in chapter 2.

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* Predecessor of the present Hochtemperatur-Reaktorbau GmbH

The THTR (Fig. 2) is being constructed in Hamm/Uentrop as a prototype reactor with a net electrical power of 3oo MW. It was ordered by Hochtemperatur-Kernkraftwerk GmbH (HKG), an association of six utilities. The contractor for the turn-key nuclear power plant ist the consortium THTR; it is composed of Brown,Boveri & Cie Mannheim (BBC is the head of the consortium and is responsible for the secondary plant), Hochtemperatur-Reaktorbau GmbH (formerly BBK and responsible for the primary plant) and NUKEM (responsible for the fuel elements). Since the erection of the THTR is of great importance to the HTR reactor line, the Federal Government contributes quite considerably to the financing of this project.

The following purposes are pursued by the THTR:

- Gaining experience with design, construction, and operation of HTR-specific components. Proof of feasibility, reduction of first-of-its-kind uncertainties.

- Gaining experience with licensing procedures for a power reactor of a new reactor line.(The THTR is subject to the same licensing requirements as the established LWR's).

- Demonstrating the alleged HTR advantages by its operation.

Contractual delivery time started on February 1, 1972 with a delivery time of 61 months. The delay and its reasons will be discussed in chapter 3.

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2. AVR Operating Experience

2.1 Brief description of the experimental reactor

The heat source of the AVR is a graphite-moderated and helium-cooled reactor core consisting of a loose bed of approximately loo ooo fuel elements (Fig. 3). These spherical fuel elements contain the fuel as coated particles (Fig. 4). The kernel of the coated particles, having a diameter of a few tenths of a millimeter, consists of uranium and thorium carbide or oxide, respectively; this kernel is coated by several pyrocarbon layers retaining the fission products. The coated particles are embedded in a graphite matrix forming the inner part of the fuel element. The outer graphite shell is free from coated particles.

During reactor operation the fuel elements are continuously added to the core, extracted through the fuel element discharge pipe, and are repeatetly circulated through the reactor core, until the full burn-up is reached (approx. 17o GWd/t HM on an average).

The helium which is heated from 275°C to 95o°C in the reactor core, enters a steam generator transferring its heat to the secondary circuit and is then recirculated to the core by two blowers.

All components of the primary circuit are accommodated in a double-walled gas-tight reactor pressure vessel. The main design data is listed in Table 1, further details are given in reference /1/.

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2.2 Availability

The time availability from start of operation until the end of 1976, has been 78 % which is remarkably high for an experimental power plant (Fig. 5). This value includes the lower availability which had to be expected for the initial phase as well as shut-downs due to the experimental program carried out with the AVR.

The relatively low availability during the fist two years primarily resulted from the initial difficulties caused by the novel characteristics of the system and the deficiency of some components, such as diaphragm compressores and solenoid valves. It can be seen in Fig. 6 that the number of unintentional shutdowns due to failure rapidely diminished with increasing operational experience and that the number of shutdowns was increasingly determined by the experimental and demonstration program.

The reduced availability in 1974 is primarily due to requirements from the licensing authorities resulting from the increase of coolant gas temperature. On this occasion the safety standards of the plant where adapted to the more stringent requirements.

2.3 Coolant gas activity

The coolant gas activity of an HTR primary circuit is low for the two main reasons: the fission products are safely enclosed already within the coated particles and the activation of the few coolant gas impurities may be neglected. Thus it is not surprising that the AVR has extremely low coolant gas activities, although a great number of fuel element types are being tested in the reactor, which partially showed a higher fission product release because of their experimental characteristics.

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The coolant gas activity has steadily decreased since the beginning of operation until early in 1975, when it reached a level of $148.1o^{1o}$ s^{-1} (4o Ci), which means a specific value of $3.3 . 1o^{1o}$ s^{-1}/MW (o.9 Ci/MW) related to the thermal power (Fig. 7).

In 1976 a particle failure occurring in a small fuel element test batch, resulted in a coolant gas activity increase to a maximum of $592.1o^{1o}$ s^{-1} (16o Ci). The activity was reduced to $222.1o^{1o}$ s^{-1} (6o Ci) again, when the temperature load of these fuel elements was reduced by circulating them into the cooler region of the reactor core and by extracting part of these fuel elements.

2.4 Activity release

The low activity in the coolant gas finds its correspondence in the low release to the environment: The mean activity release to the atmosphere by nobel gases per year is about $74 \times 1o^{1o}$ s^{-1} (2o Ci) and by tritium about $92.5 \times 1o^{1o}$ s^{-1} (25 Ci). This results in an radiation exposure at the point of maximum load in the environment which is $1o^{-5}$ mJ/kga ($1o^{-3}$ mrem/a) compared to the natural radiation exposure of 1.1 mJ/kga (11o mrem/a). The release of activity carried by aerosols is about $o.74 \times 1o^{1o}$ s^{-1} (2oo mCi/a). The exposure of the environment resulting from this activity may be neglected. The release of radioiodine is so low that it remains below the limit of traceability. The activity occurring in liquid form is about $222 \times 1o^{1o}$ s^{-1} (6o Ci) per year.

2.5 Radiation exposure of personnel

In correspondence with the low activity also the radiation exposure of the personnel is very low.

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This applies to the plant personnel and, in particular also to the external personnel. The radiation exposure has steadily decreased over the years. The integral dose for all persons professionally exposed to radiation was about 45o mJ/kg (45 rem) in 1976, of which only o.8 mJ/kg (o.o8 rem) comes on external personnel. The mean radiation exposure of all persons exposed to radiation was 3.4 mJ/kg (34o mrem) per person in 1976. This mean value is expected also for 1977, although repair work was done on parts of the fuel circulating system implying a particular radiation hazard.

The low radiation exposure of the operating and maintenance personnel is a typical HTR characteristics which is of particular interest to utilities.

Reference is made to literature reference /1/ for further information especially on the time curve of activity release and radiation exposure.

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3. Experience Gained with the Construction of the THTR

3.1 Brief description of the prototype reactor

Also in the THTR spherical fuel elements are used (Fig. 8). The components of the primary circuit are also integrated in a reactor pressure vessel, which however, in contrast to the AVR, is designed as a prestressed concrete reactor vessel (PCRV). The central cavity accomodates the reactor core surrounded by six steam generators. One circulator is attributed to each of steam generators. The helium flows downward through the core and is being heated from 260°C to 765°C, thus supplying conventional steam conditions on the secondary circuit.
The reactor is controlled by absorber rods, which can be freely moved in bore holes in the side reflector. These so-called reflector rods also represent the first shutdown device. For long-term shutdowns in-core rods are directly inserted into the pebble bed by pneumatic drives.

The THTR main design data and their differences towards the AVR data are listed in Table 1.

3.2 Present state of erection and assembly

By the end of 1977 the construction status is determined by the fact that the PCRV is completed, its essential internals have been mounted, and manufacture of the components of the primary circuit virtually completed (Tab. 2).

Only some residual work remains to be done on the thermal barrier of the steam generator penetrations.

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Assembly of the internals has been completed with the exception of an access opening left for assembly, and suspension of the top reflector and the thermal top shield. The six steam generators are being manufactured in parallel with a certain phase lag to each other. Delivery on site will be spaced from spring till the end of 1978. The coolant gas circulators are stored by the manufacturer until their delivery on site (autumn 1978).

The fuel elements, the burnup measurement reactor, the fuel circulating system, and the shutdown facilities are also completed and ready for use in the reactor or assembly on site.

As can be seen from Table 2, also the dry-cooling tower, the reactor hall, the turbine building as well as the reactor service building and the reactor operation building are largely completed.

The future activities are concentrated on the peripheral plants and the assembly of the secondary circuit. As can be seen in Fig. 9, the critical path of the remaining time schedule consists of

- completion of the gantries above 29 m and the annular compartments on top of the PCRV
- assembly of steam generators and the HP-steam pipes
- delivery and assembly of further auxiliary circuits

Mainly the German licensing procedure has caused the delay in the construction progress.

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3.3 Present state of licensing procedure

In the Federal Republic of Germany the license for the erection of nuclear power stations follows the progress of construction by so-called partial construction permits. The licensing procedure for the THTR was initiated on the 12th of January, 197o. 8 partial construction permits have since been granted. (Table 3).
The reactor concept was licensed among other items by the first partial construction permit.

The remainder of the partial construction permits is expected to be granted in 1978, the main items to be licensed being the steam generators, the circulators, the secondary plant, the absorber rods, and the instrumentation and control.

The operating license is scheduled for 1979.

The licensing procedure is based on the Federal Atomic Law, which requires to take into account the status of science and technology even during construction. This has resulted in various supplements and amendments which had to be taken into account afterwards.

3.4 Experience gained

3.4.1 Licensing procedure

In agreement with authorities and experts originally the duration for a licensing procedure had been assumed to be approximately 10 months. In reality the period of engineering, examination, and expertizing until

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granting of the construction permit took about 15 months for the buildings, and up to 36 months for some mechanical components.

The main reasons for these substantially longer periods until the granting of the partial construction permit are,

- the more stringent licensing requirements, following the new status of science and technology,
- the consequent application of the LWR licensing procedure to the prototype reactor of a new and different reactor line.

The most important licensing requirements which have subsequently become more stringent, are summarized in Tab. 4. They partly resulted in substantial re-design, required comprehensive verifications and interfered with the manufacture and assembly of the components and the erection of the buildings, thus resulting in considerable delays and cost increases.

As an example, the consequences of the requirement of coping with spontaneous tube ruptures (guillotine rupture) led to a delay of 29 months. As a countermeasure very many pipe restrainers had to be mounted to avoid consequential damage and possible chain effects. For this purpose a particular development and testing of shock absorbers as well as re-designing of the upper gantry and were necessary.

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Contractor and utility had pleaded already at an early date that today's application and execution of the atomic law licensing procedure, currently in use, is not appropriate to prototype reactors for two reasons:

- The licencing procedure developed from the licencing practice of the LWR's does not take into account the HTR characteristics. Thus, e.g., in the THTR, even in the hypothetical event should all heat removal systems and the scram system fail, several hours are available to take proper internal and external counter-measures.

- It is only the construction and operation of a prototype reactor which provide the experience whose evidence is being claimed already in the licensing procedure. In addition, the deficiencies recognized during construction and operation of the prototype will be avoided in the reactor line, thus the failures occur only in the individual prototype plant.

Industry has not only prepared THTR design criteria in cooperation with the licensing authorities; it also has recently proposed to supply the verifications for the prototype, if need arises, in a later phase of the licensing procedure and during commissioning, in particular if long-term effects are concerned.

Since the main efforts in the erection of the THTR have now been transferred to peripheral plants and to the secondary circuit, industry expects that the slogan "build and test", which is appropriate to a prototype, may have a better chance, since these components are accessible and exchangeable. Thus the remaining construction could be considerably accelerated.

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The delays due to the licensing procedure mainly resulted from the software problems (changes in procedures, verifications, etc.) and less from changes of hardware. Hence, if the present stringend licensing requirements are taken into account in the design of a follow-up plant right from the start, and as experience and verifications are drawn out of the THTR, similar delays as encounterd in the THTR erection will not occur. Further more, industry intends to design follow-up plants in principle to a still further reduced residual risk to meet possibly increased requirements in advance.

3.3.2 Manufacture and assembly

Difficulties arising in the course of manufacture and assembly were not caused by the specific HTR-technology, but by such problems arising also in conventional technology. There, however, they would in many cases never be detected, since the extent of testing in no way corresponds to the testing procedures applied for the construction of the THTR.

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Thus a total of almost 5 millions of individual tests have been carried out by the end of 1977, (material tests, non-destructive tests, dimension checks, etc.), whose results are documented in approx. loo ooo certificates. Just on the liner and the penetrations of the PCRV about 500 000 individual tests were carried out and documented.

The extremely low reject or repair rate observed during these tests, shows a very high standard of manufacture held by the subcontractors chosen. However, the good result can only be achieved when engineering procedure tests are carried out prior to the manufacture under largely realistic conditions.

The contact with subcontractors always ran smoothly, when the necessary research and development work as well as the design work was completed before placing of the order and, hence, detailed requirements, specifications, and boundary conditions could be provided in good time.

Nevertheless, the number of subcontractors ensuring such a high quality standard as required for nuclear power plants is limited. In particular foreign suppliers of components created some problems. The licensing procedures practiced abroad are substantially different from the testing and acceptance requirements and their documentation as used in the Federal Republic of Germany. Hence, foreign companies had to go through a lengthy and expensive learning process in order to meet the German standards.

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4. Conclusions

The construction and operation of the AVR confirmed the favourable characteristics of the High Temperature Reactor, and many of the expectations have even been surpassed. This experimental reactor not only furnishes excellent operating results regarding availability and radiation exposure of environment and personnel, but in addition, it has been possible to gain valuable experience to the HTR line. In the first place, the fuel element should be quoted which was tested in a large-scale test under HTR-specific conditions. Also for other components the more than 10 years of operation have furnished useful knowledge.

Although the coolant gas outlet temperature had initially been designed to 850°C, it was increased to 950°C in February 1974. The several years of AVR operation at this maximum coolant temperature ever achieved in a nuclear power plant demonstrates the applicability of the HTR for the generation of nuclear process heat. In addition, the favourable safety behaviour of the AVR was strikingly demonstrated by the so-called rod jamming test: The coolant circulation was interrupted at full power and the insertion of all absorber rods was prevented. The reactor scrammed inherently, remained sub-critical for one day, and then stabilized at a power in the kW-range.

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It can thus be summarized that the AVR can be considered as an indispensible and extremely successful experiment, which has advanced the HTR development by a decisive step.

The construction of the THTR made evident the strong influence of the licensing procedure on the erection sequence. Due to subsequent and more stringent licensing requirements numerous re-designs, verifications, and tests became necessary causing considerable delays in manufacture and assembly. According to the present time schedule the delivery time is 108 months. This has lead to a total delay of 47 months, out of which only 9 months resulted from delayed supply of shutter tubes. The other delays are directly or indirectly caused by the licensing procedure. This caused also about 90 % of the overcost.

It is remarkable that the HTR characteristics offered sufficient flexibility to meet the more stringent safety requirements.

The consequent application of the LWR orientated licensing procedure, to the prototype of a new reactor line is the main reason of the delay incurred, however, it probably solves part of the problems in follow-up reactors right from the start. Based on the experience gained from the erection of the THTR it can be expected that the construction period of follow-up plants can be reduced to an acceptable extent.

In the Federal Republic of Germany the development of the High Temperature Reactor has been given a new direction. In this program the THTR holds a key position and is thus being sponsored with priority by the Federal Government and all parties involved in the HTR development.

Literature

/1/ 10 Jahre Stromerzeugung mit dem Kugelhaufenreaktor der AVR, AVR/HRB Druckschrift, 17.12.1977

Nuclear Power Plant	AVR	THTR
Overall Plant:		
Electric Power Net Output, MW	13	300
Recooling Mode	wet	dry
Reactor Core:		
Thermal Power, MW	46	750
Mean Power Density, MW/m^3	2.6	6
Height/Diameter, m	2.47/3	6/5.6
Number of Fuel Elements	98000	675000
Helium Pressure, bar	10.9	39
Absorber Rods In-core/Reflector	-/4	42/36
Steam Generator:		
Number	1	6
Helium Inlet Temperature, °C	950	750
Superheated Steam Temperature, °C	505	550/535
Reactor Pressure Vessel:	Steel	Prestressed Concrete
Outer Diameter, m	5.8	24.8
External Height, m	24.9	25.5

AVR HRB	AVR and THTR Design Data	77.26-4

Table I

	State of Completion %	
	in Workshop	on Site
Prestressed Conrete Reactor Vessel	–	100
Internals	100	90
Fuel Elements	100	–
Burn-up Measurement Reactor	100	–
Fuel Circulating System	80	10
In-core Rod Assemblies	90	–
Reflector Rod Assemblies	55	–
Steam Generators	80	–
Main Circulators	95	–
Main Turboset	100	10
Water/Steam Circuit	10	–
Buildings	–	90
Dry-cooling Tower	–	100

HRB | State of THTR Erection End 1977 | 77.26-7

Table II

Table 3 THTR Partial Construction Permits

	Components concerned,	granted
1	Reactor concept, reactor building site permit Foundations and annular support wall of PCRV Cranes and elevators in reactor building	3/5/71
2	PCRV, intermediate floor in annular room below PCRV Hoists in reactor building	18/8/72
3	Electrical equipment building	8/1/73
4	Engine building, burn-up measurement reactor	7/9/73
5	Fuel circulating system, gas circuits, storage container for fresh fuel absorber and moderator elements without instrumentation and control system	19/11/74
6	Metal and ceramic	4/3/75 17/4/75
7	Electrical installations Auxiliary cooling water systems	22/6/76
8a	Installation of condensate purification plant for water/steam circuit	16/7/76

8b	Cooling water circuits	1978
9a	Steam generators, circulators, secondary plant	1978
1o	In-core and reflector rods, control gas and NH_3-system, VGD-S, rod protection floor	1978
11	Instrumentation and control system: package 1	1978
12b	Instrumentation and control system: removal of decay heat, reactor protection system, ventilation system, overall control steam generators, water steam circuit, activity control.	1978

Table 4 THTR Safety Requirements Fulfilled Subsequently

No.	Safety Requirements increased subsequently	Year	Measures taken
1	Protection against airplane crash (Starfighter)	1972	Local separation of all redundant safety-related components. Securing of reactor hall gantries against torsion. Splitting-up of electrical installations building
2	Protection against shock-wave	1972 and 1976	Bunkering of safety-related important components. Verification of compatibility in the reactor hall.
3	Protection against earthquake (vertical acceleration)	1976	Re-calculation of the buildings already licensed. Design of all components on the basis of corresponding verifications
4	Object protection	1973	Planning of comprehensive technical and administrational measures
5	Protection against spontaneous pipe rupture (guillotine rupture)	1975	Pipe restrainers, in particular on all pipes in the reactor hall
6	Protection against spontaneous vessel failure	1976	Verification of permissible damages, removal of individual vessels

...

No.	Safety Requirements increased subsequently	Year	Measures taken
7	Assumption of failure of components not examinable by in-service inspection	1974	Verification of permissibility of failure, otherwise additional design measures
8	BMI-Criteria, RSK-Criteria	1974	Establishment of new design criteria applying BMI-Criteria in an analogous manner
9	Single failure criteria	1974	Application of these criteria including the event of repair. Changes in the overall decay heat removal and emergency power supply systems. Making use of inert and favorable safety behaviour of HTR
lo	KTA Regulations		Are being taken into consideration upon their publication
11	New Radiation Protection Regulations		Effects cannot be fully appreciated yet. Reduction of radiation level in operating rooms and corridors to o.5 mrem/h.
12	Reliability analyses, diversity of plant protection trigger	1976	Redesign of decay heat removal system and trigger system
13	Quantitative verification of inherent safety characteristics		Verification that the THTR may be left without any heat removal at all for several hours, and then heat removal is possible without permanent damage.

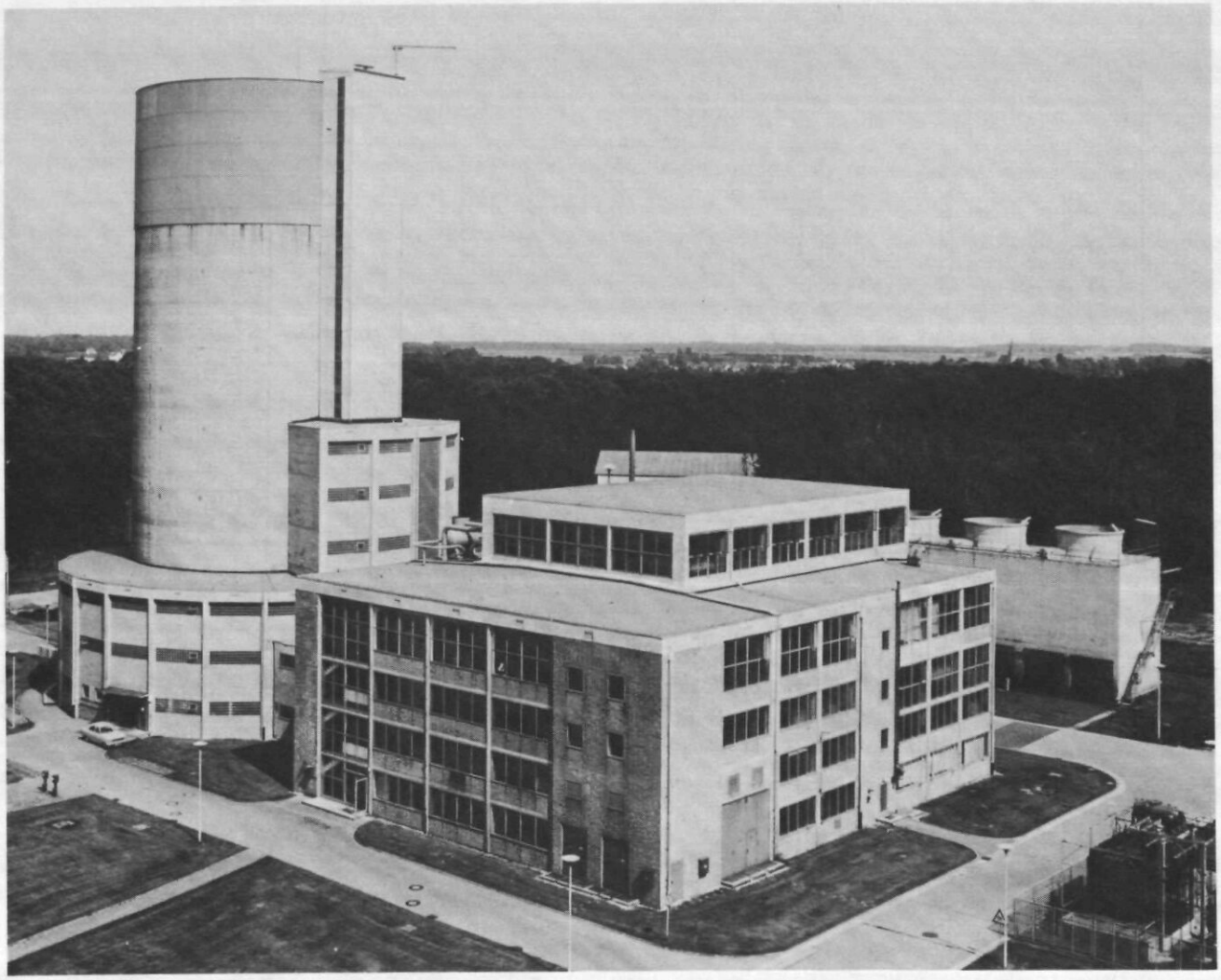


Fig. 1
AVR Experimental Reactor

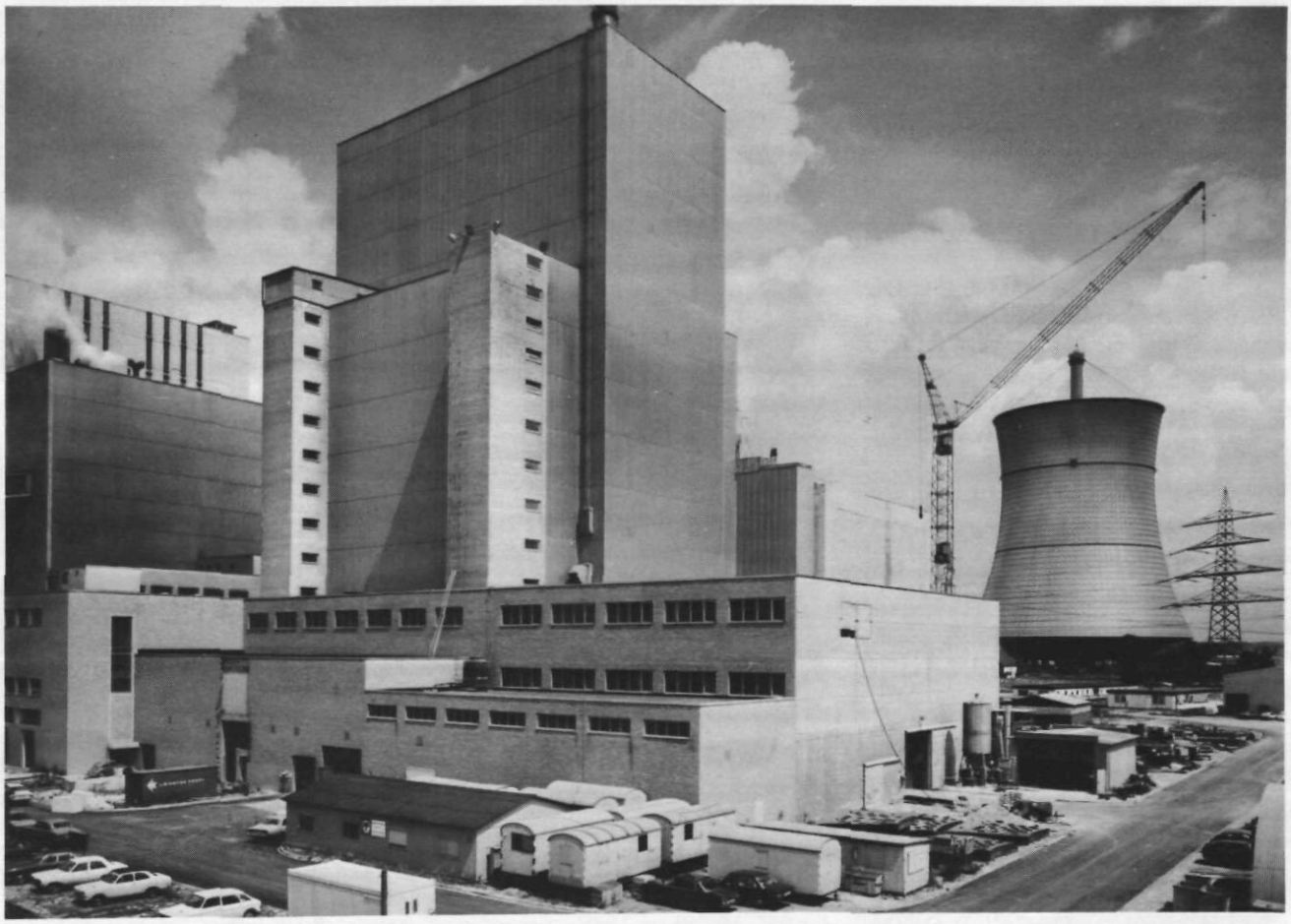


Fig. 2
THTR Site

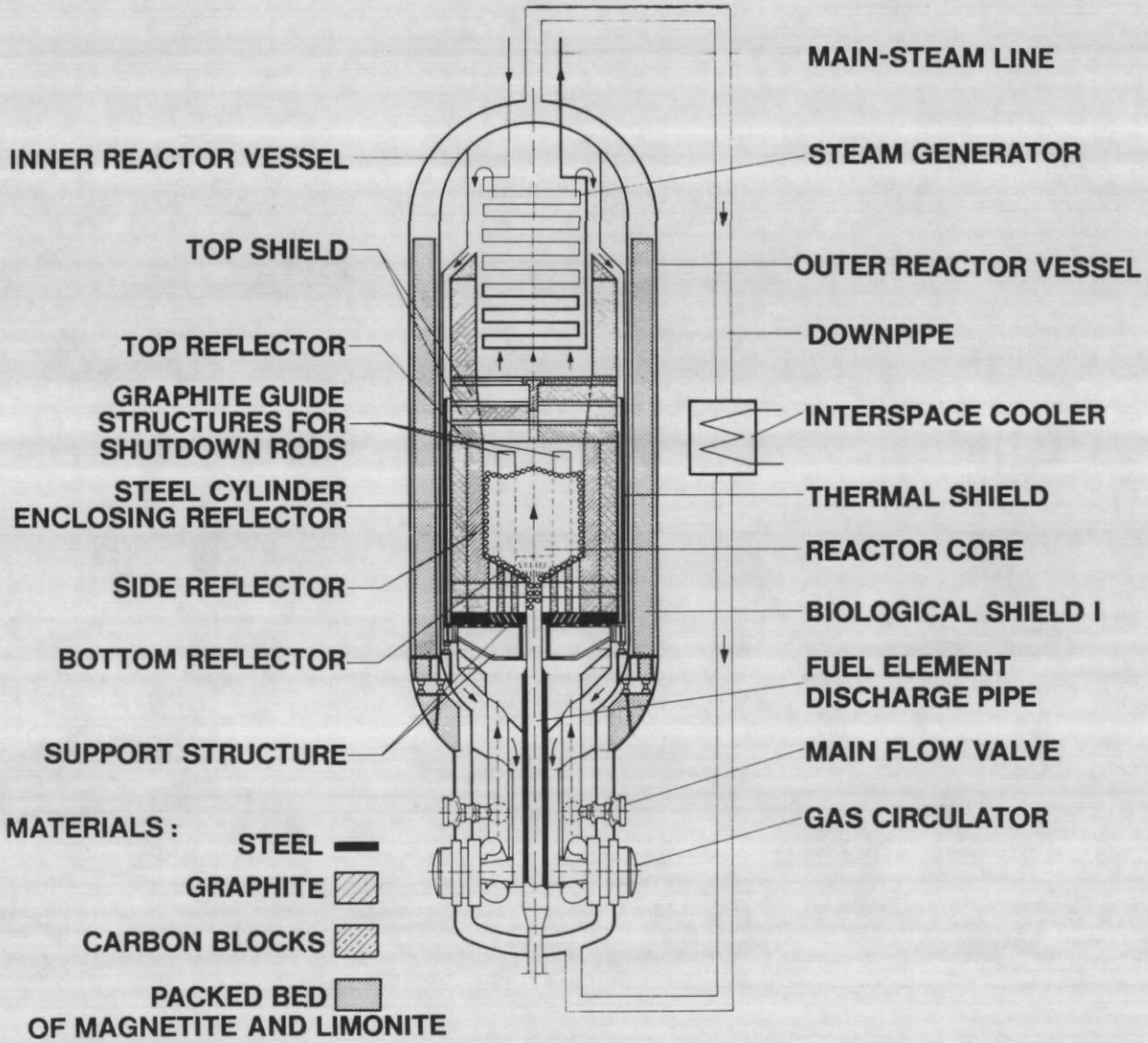


Fig. 3
Vertical Section AVR Reactor

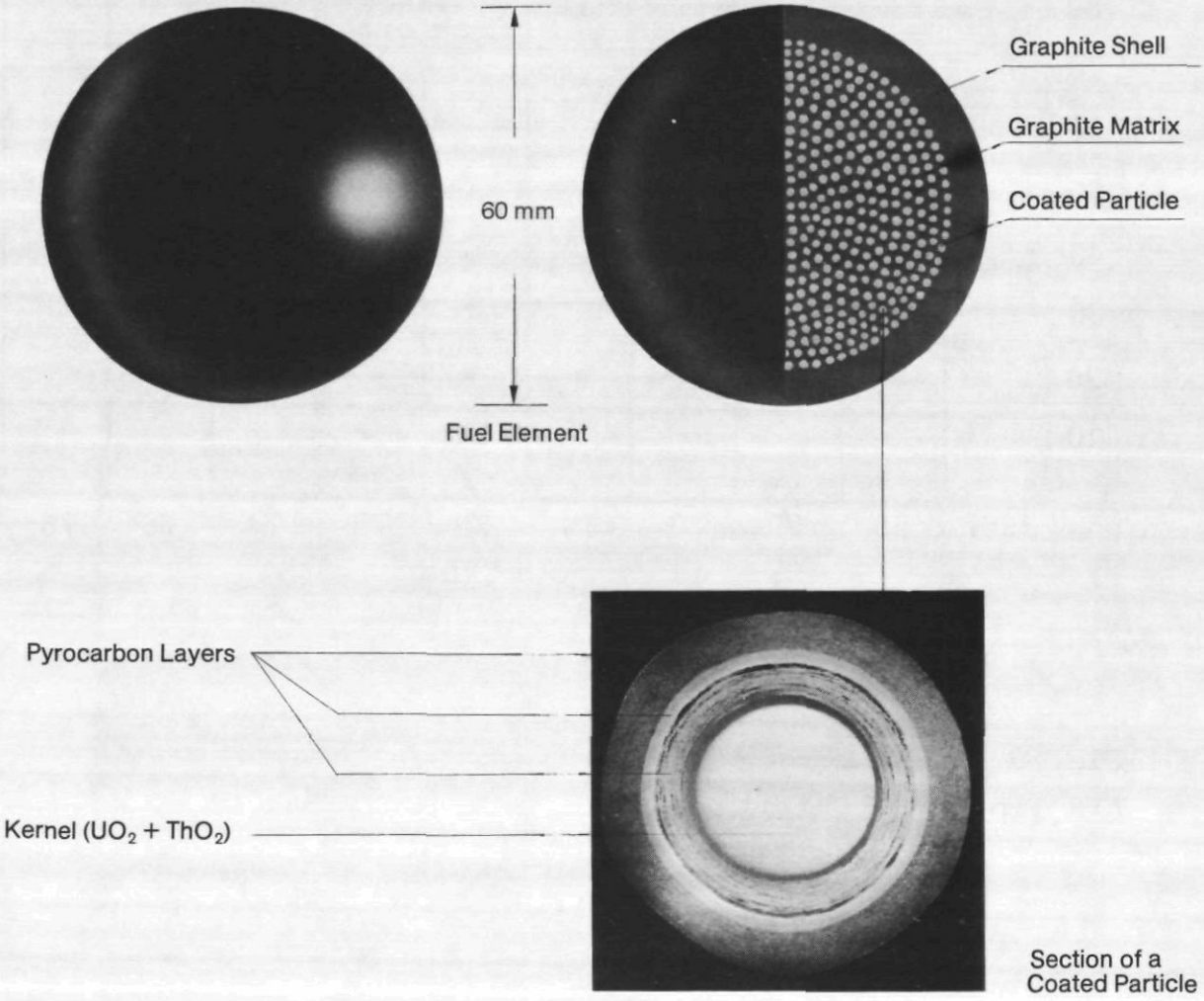


Fig. 4
HTR Fuel Element

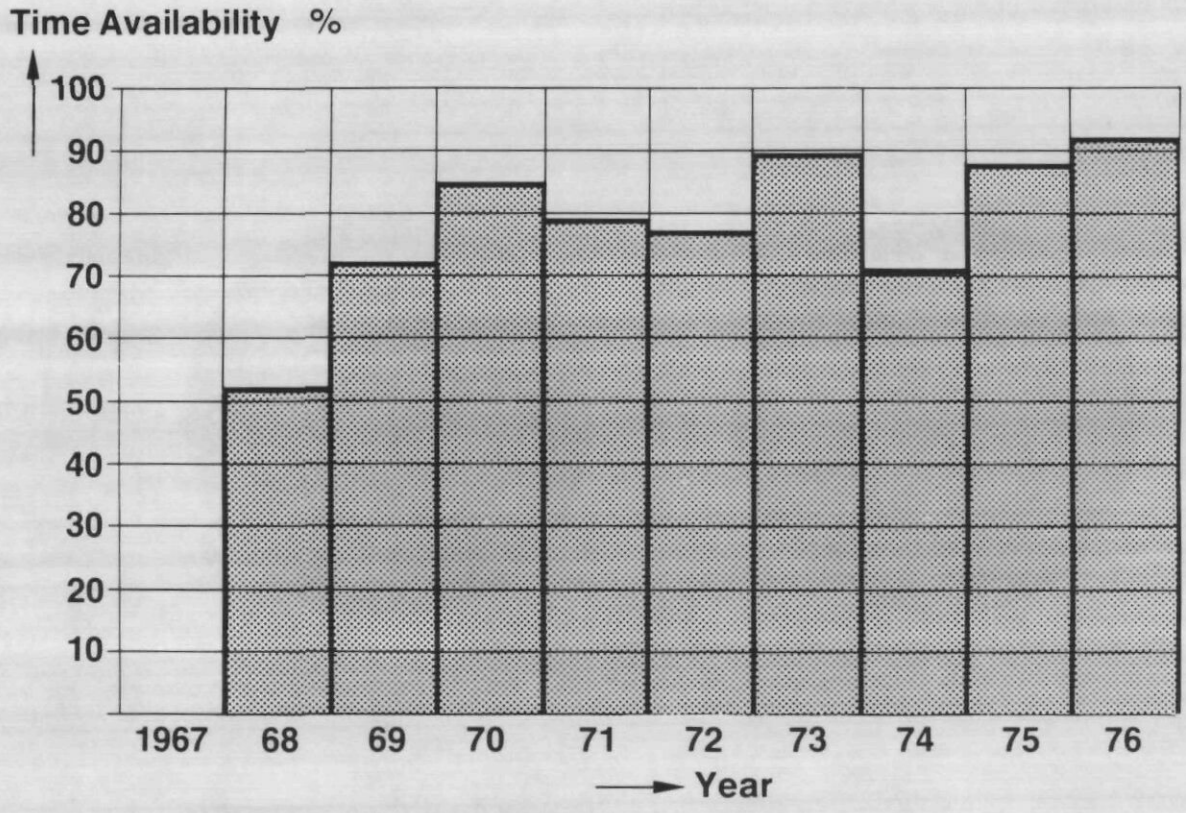


Fig. 5
AVR Time Availability

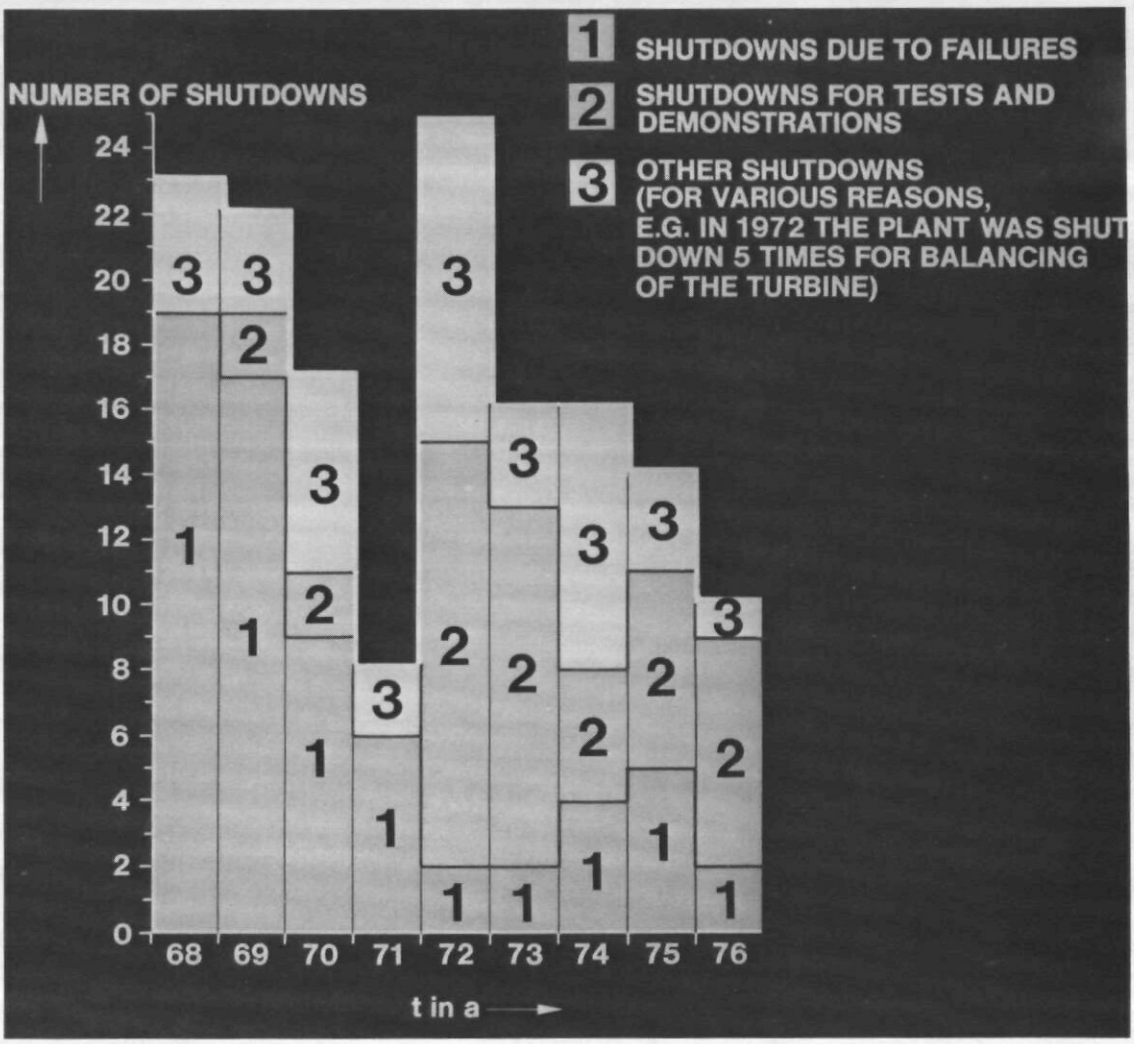


Fig. 6
Shutdowns of the AVR Experimental
Nuclear Power Plant for the Years 1968 - 1976

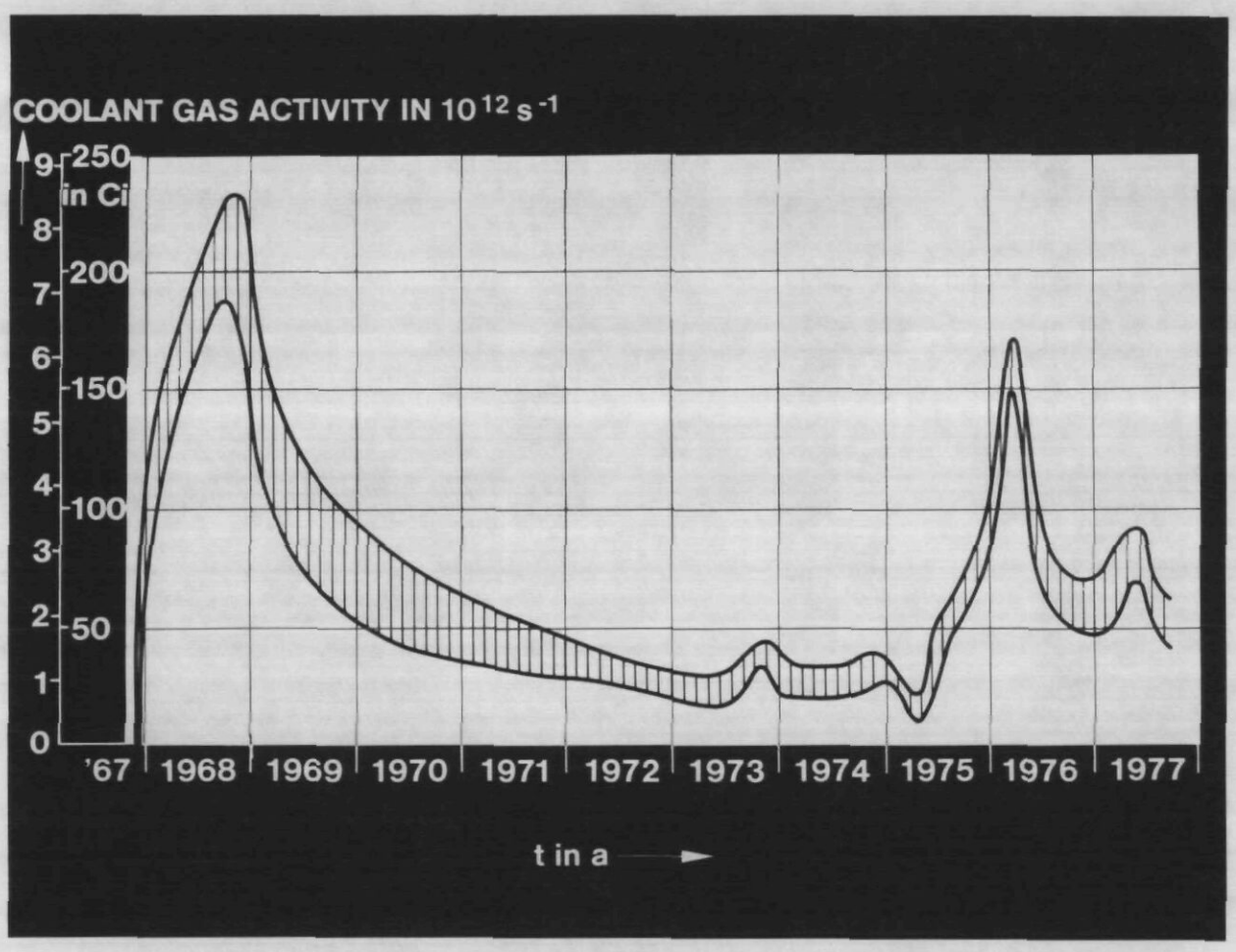


Fig. 7
Coolant Gas Activity in the AVR Primary Circuit

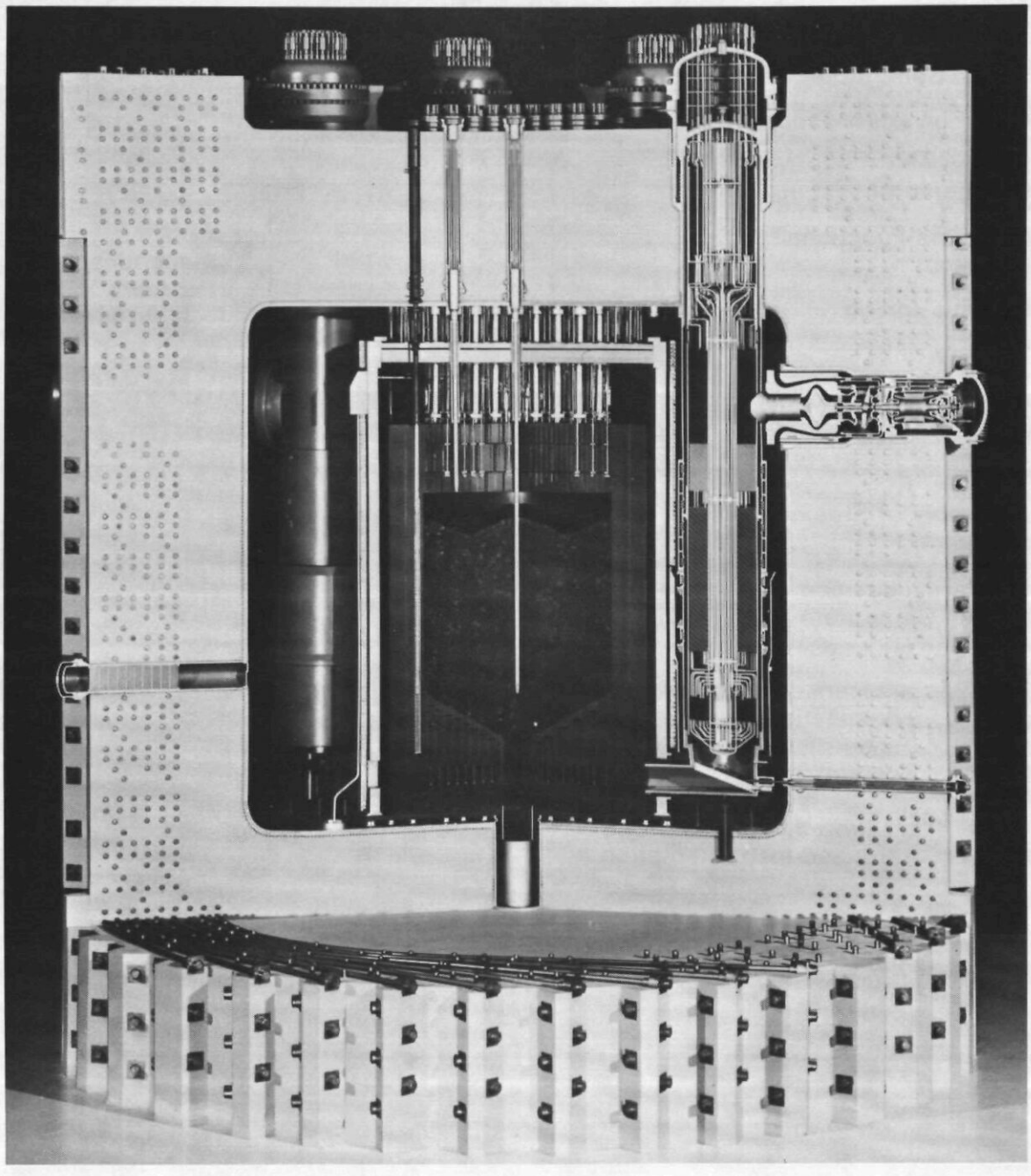
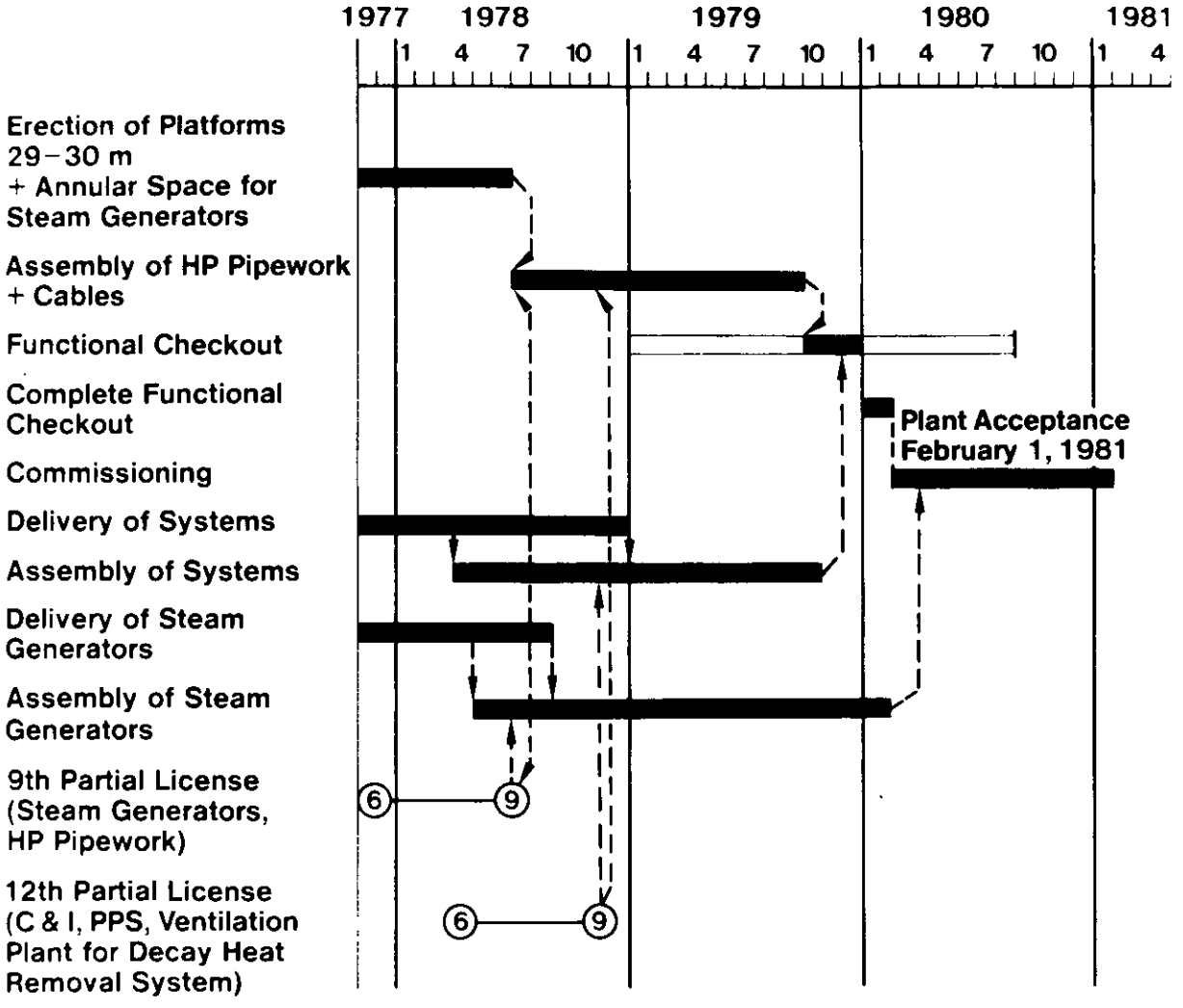


Fig. 8
THTR Reactor Pressure Vessel



PPS = Plant Protection System
C & I = Control and Instrumentation

Fig. 9

Critical Paths

HTR-Technology

The Nuclear System and its Application for Power Production and Process Heat

R. Schulten

The fundamental principle underlying the High-Temperature Reactor is the utilization of ceramic materials and helium as coolant. The most important material for the fuel elements and for the structure of the reactor core is graphite. The actual fuel is a mixture of uranium and thorium oxide, which is encased in graphite. The utilization of graphite at high temperatures needs a noncorrosive coolant. Helium was found to be the most favourable of the various possibilities. To meet the demand that charging and discharging of the reactor take place during operation, ball-shaped fuel elements are necessary for the operation of the reactor. Such fuel elements are shown in Fig. 1. The actual fuel element is a coated particle, which consists of an approx. 0.2 mm uranium/thorium-oxide kernel surrounded by a buffer zone of soft graphite. This zone is enclosed by a graphite coating, which is impermeable towards fission products. The main function of the buffer zone is an elimination of mechanical contact between the fuel oxide kernel and the graphite coating. The coated particles thus obtained have a diameter of less than 1 mm. About 5 - 10 thousand are pressed together along with graphite powder to form a fuel ball, which has a diameter of 60 mm. The outer zone of the graphite ball does not contain fuel. Depending on the composition of the oxide kernels, cycles with low-enriched uranium and with uranium/thorium of varying degrees of enrichment can be run with this fuel element. Thus it is possible to so increase the thorium content that, assuming a corresponding reprocessing technology, near-breeders with a conversion factor of 0.95 can be realized. By using a breeder blanket, which could occur in the form of graphite balls with thorium, the breeding factor can be increased to 1. In principle, fuel cycles are possible with and without reprocessing. In the latter case, the fuel elements can be burnt up to such a large extent that reprocessing is economically not necessary.

The fuel elements enter the reactor at the top (Fig. 2). They move downwards with a velocity of approx. 1 to 2 mm/d and reach the bottom of the reactor from where they are removed by discharge ducts. During this passage downwards, the fuel material is gradually burnt up whereas the concentration of the fission products increases. This loading scheme leads to a power and temperature distribution within the reactor which is represented in Fig. 3. The highest power density is at the upper "cold" end of the reactor and it decreases rapidly in the downward direction. As a result, the coolant is heated relatively quickly in the upper part of the reactor and the temperature difference is small between the temperature at the centre of the fuel elements and the helium outlet temperature in the lower part of the reactor. This it is possible to design the reactor so, that the outlet temperature differs only slightly, normally less than 150 ^{o}C, from the maximum temperature within the fuel elements. Through this characteristic charging of the reactor it is possible to attain higher outlet temperatures even up to 950 ^{o}C without an unpermissible temperature load on the fuel elements.

The power density of a reactor in the axial and radial direction is again shown in Fig. 4. Due to the varying enrichment of the fuel material in the inner and outer zone it is possible to keep the overall power, integrated in the axial direction, fairly constant. This means a steady temperature profile for the outlet helium can be realized. The temperature deviation can be expected to be $\pm$ 50 ^{o}C at the most.

Fig. 5 shows an exact representation of the reactor core, wherein the components of the THTR are shown in detail. In the figure we see the reflector, which surrounds the whole thermal shield, the thermal shield, which supports the reflector, and from which the upper reflector is suspended, as well as the steam generators, the circulators, the control rods and the lower outlet ducts for the ball-shaped fuel elements. All these components are arranged within a prestressed concrete vessel which is gas-tight and insulated on the inside.

Fig. 6 shows the size of the AVR reactor, the THTR and a process heat reactor of 3000 MW. The AVR and THTR have only one ball outlet duct whereas 6 such ducts are necessary for the large reactor, in order to attain a sufficiently uniform velocity distribution of the fuel elements during their passage through the reactor core.

An important question is the release of radioactive fission products from the reactor core. Irradiation and checking-up programmes are being carried out since more than 10 years. For the production of nuclear process heat, the maximum temperature of the coated particle in our reactor is approx. 1150 °C. Most of the irradiation was, therefore, carried out at about 1200 °C. The mass tests of fuel elements in the AVR reactor were especially valuable. The diffusion behaviour of cesium, silver and noble gases were investigated during the checking-up. Furthermore, the amount of coated particles which break during fabrication or during their lifetime in the reactor were determined. An important factor for the behaviour of the fuel elements is also the contamination by uranium and thorium, which accumulate on the outside of the coated particles during the fabrication process. Upto now so-called H.T.I.-particles, which are also utilized in the THTR, have been particularly investigated. Their characteristic is the high production temperature of approx. 1600 °C. So-called L.T.I.-particles, which precipate at low temperatures of about 1200 °C, were also investigated. Furthermore, special importance was attached to a development, which aims at chemically binding cesium and other radioactive fission products by additives of aluminium oxide and silicium oxide, thus diminishing diffusion from the particle kernel. Based on the present results of investigations, a broad calculation model was developed, which can be used to predict the behaviour of the particles and the fuel elements.

Such results are represented in Fig. 7 and it can be seen that the L.T.I.-particles, with and without additives, can hardly guarantee the conditions for nuclear process heat regarding the retention of fission products. However, the above-mentioned addi-

tives can apparently improve the retention of cesium by more than a factor of 100. The remaining dash curves show the behaviour of the H.T.I.-particles. From the figure it can be seen that for a desired lifetime of approx. 900 days, these particles can attain a release factor of 2×10^{-3}, taking into consideration the contamination and the breakage. In the presence of additives under the same considerations as mentioned above, a retention capacity for the particles of less than 10^{-4} can be achieved.

From these results it can be inferred that it will be possible to produce particles and fuel elements with a sufficiently small release of fission products for future process heat applications at 950 °C. In future programmes, it is aimed at improving the behaviour of the particles further by a silicium carbide coating, so that the release of radioactive silver into the circuit is also decreased. There are programmes analogous to this calculation model, with which the behaviour of the radioactive substances into the primary circuit can be predicted. The results of these caluclations have been tested and checked on the behaviour of the AVR reactor.

In recent years, the feasibility of large pebble-bed reactors having a power of 3000 MW has been verified by various studies. Fig. 8 shows a process heat reactor of 3000 MW and a helium outlet temperature of 950 °C. The reactor, the steam reformer for the conversion of methane and the steam generator are placed successively. The cooled helium is led concentrically back to the reactor by the circulator. In these studies it was decided to place all the components of the primary circuit in an integrated system in a prestressed concrete vessel. This decision was reached mainly due to technical safety reasons to exclude bursting of the primary circuit right from the beginning.

With this reactor a plant with a gas turbine has also been designed (Fig. 9). Here also the reactor, the gas turbine, the recuperator, the cooling system and the after-heat removal system are placed in a prestressed concrete vessel. For a helium outlet temperature of 850 °C, an efficiency of 41 - 42 % can be

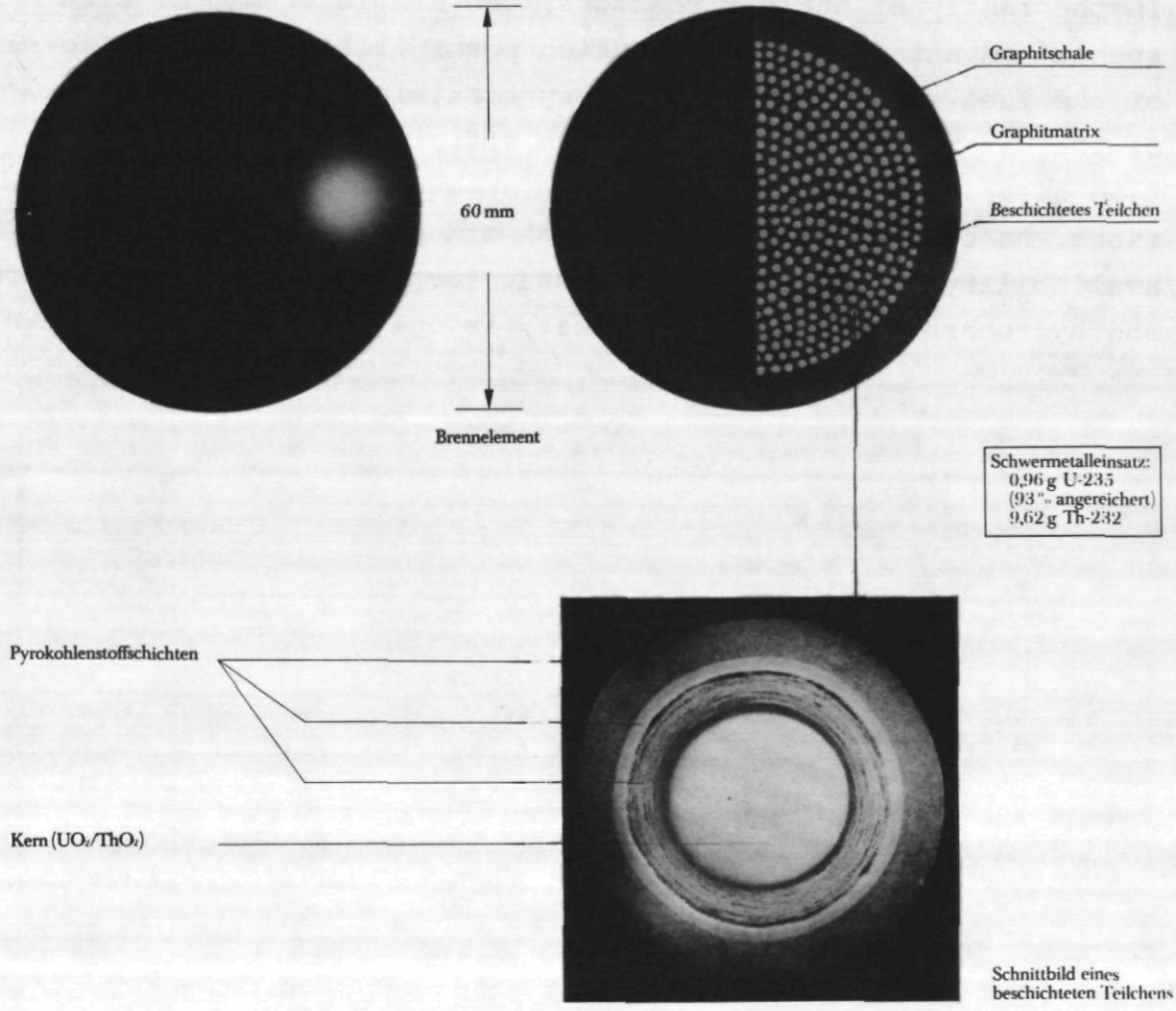

60 mm
Brennelement
Graphitschale
Graphitmatrix
Beschichtetes Teilchen
Schwermetalleinsatz:
0,96 g U-235
(93 % angereichert)
9,62 g Th-232
Pyrokohlenstoffschichten
Kern (UO_2/ThO_2)
Schnittbild eines
beschichteten Teilchens

Fig. 1

attained. It it is possible to improve the turbine materials to withstand a temperature of 950 °C, an efficiency of 44 % can be expected. The waste heat of the gas turbine power plant at 120 °C can allow district heating for the heating of private houses practically free with this system. The main problem of this concept is to keep the contamination of the turbine as low as possible, so that maintenance and repairs of the gas turbine after possible decontamination are sufficiently easy.

In the family of nuclear reactors, the HTR is a reactor with special advantages and application possibilities. The choice of the fuel element cycles is very flexible. The application of the gas turbine leads to a high efficiency and at the same time district heating is available at extremely economic conditions. An especially important area of application, namely the availability of nuclear process heat for the production of hydrogen and hydrocarbons with nuclear heat, is only possible with the HTR.

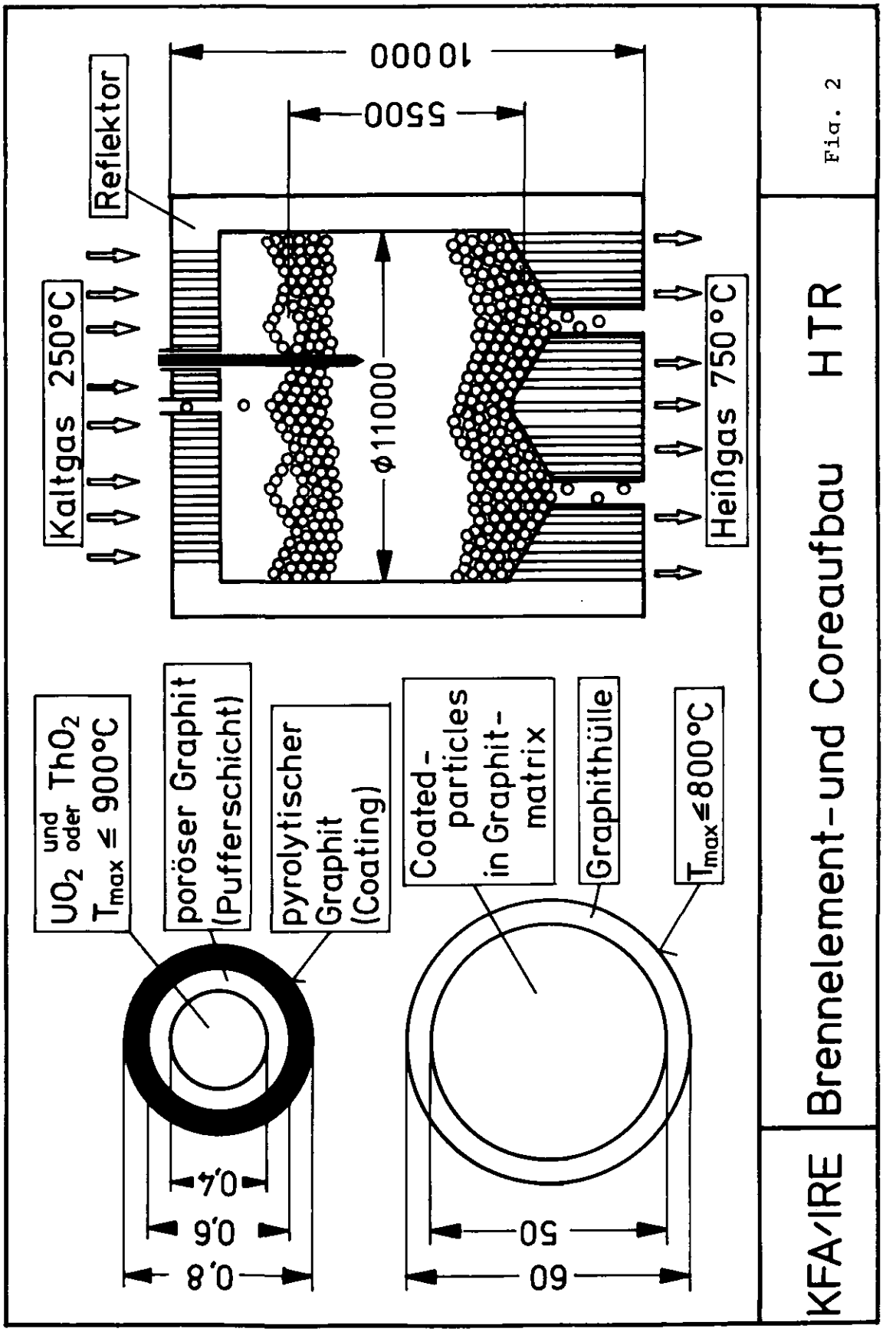
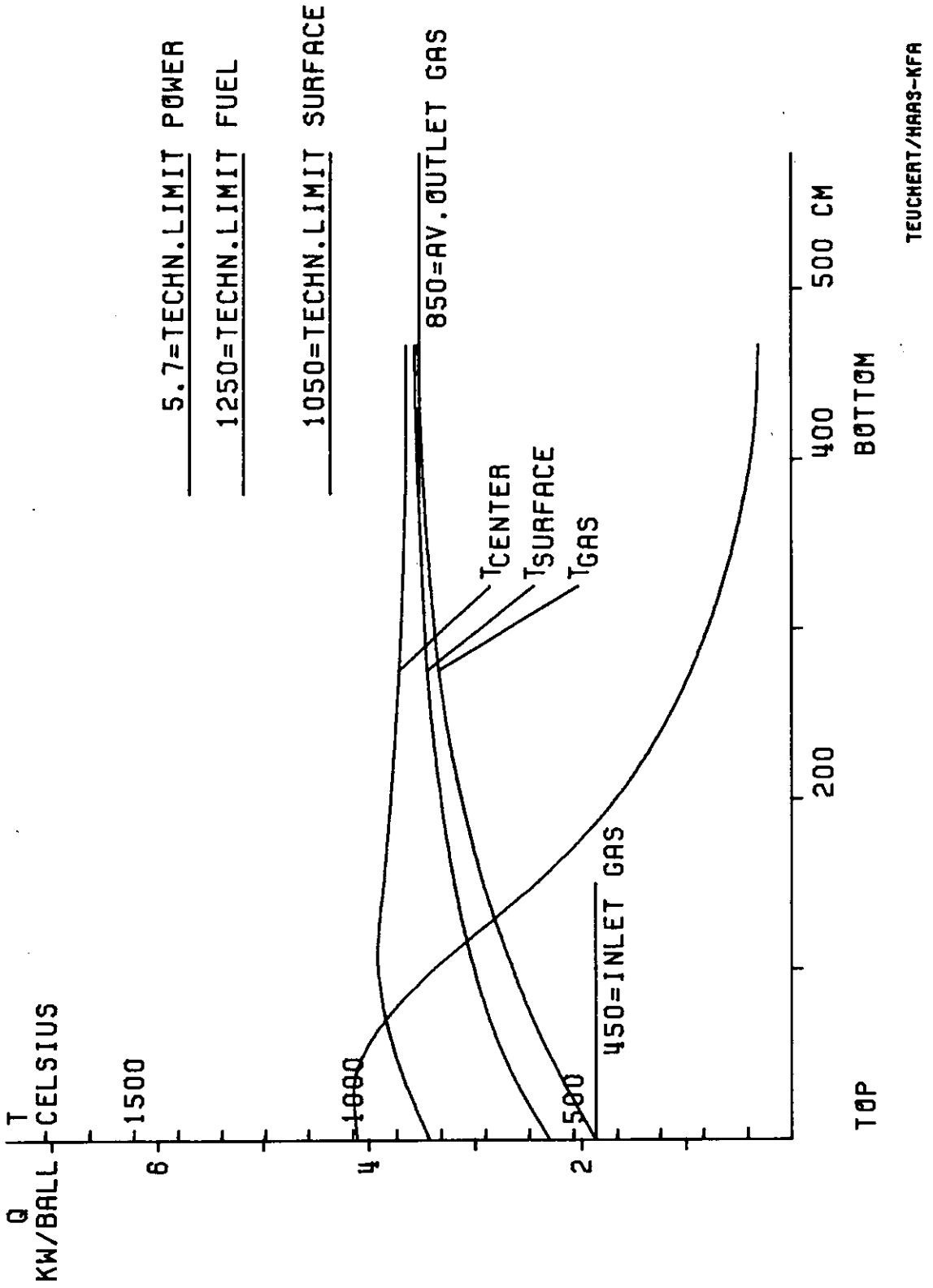

KFA/IRE
Brennelement-und Coreaufbau HTR
Fig. 2
Reflektor
10000
5500
Kaltgas 250°C
Heißgas 750°C
ϕ11000
UO2 und oder ThO2
Tmax ≤ 900°C
poröser Graphit
(Pufferschicht)
pyrolytischer
Graphit
(Coating)
Coated-
particles
in Graphit-
matrix
Graphithülle
Tmax≤800°C
0,4
0,6
0,8
50
60

Fig. 3: AXIAL PROFILE OF POWER AND TEMPERATURES



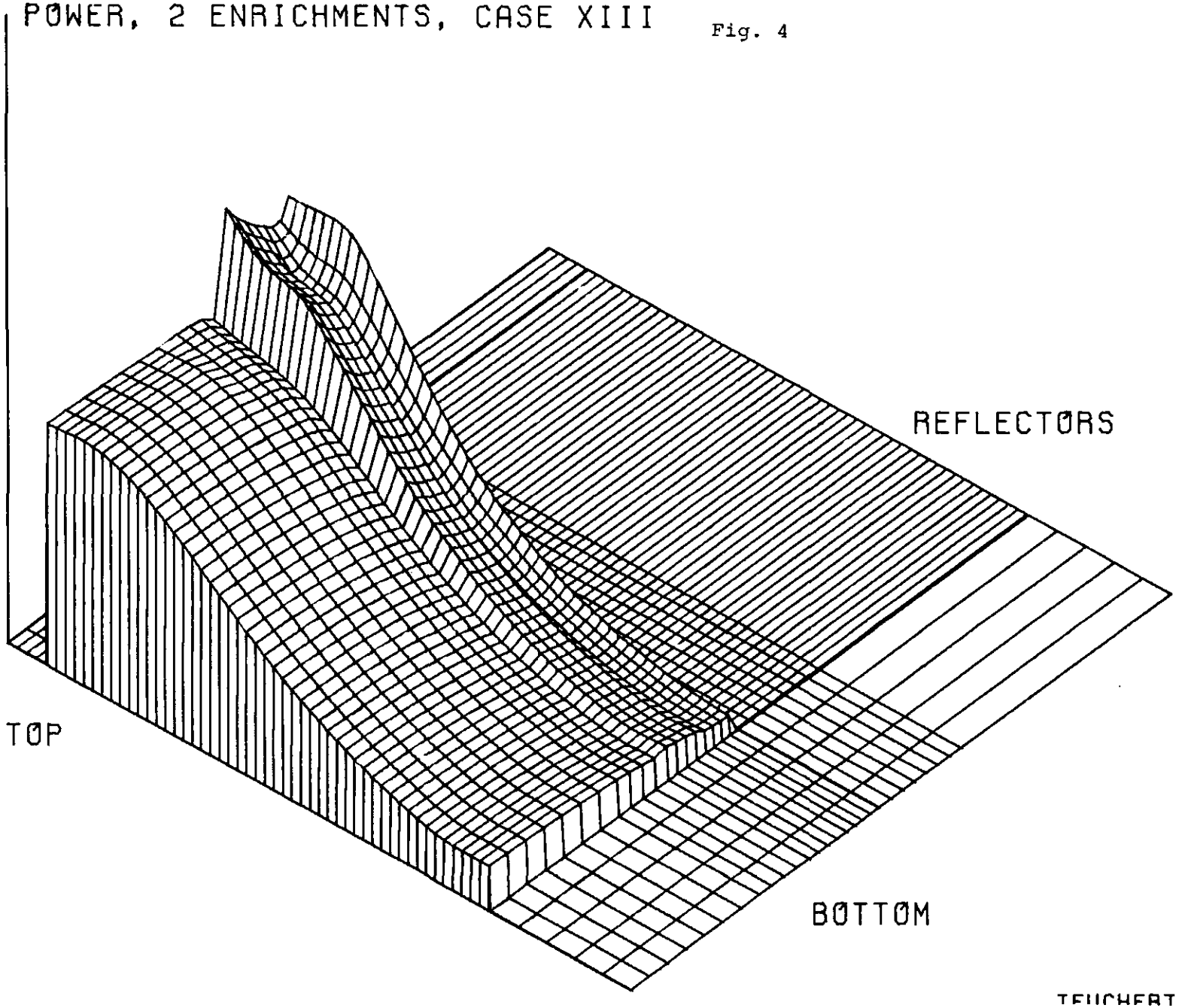


Fig. 4

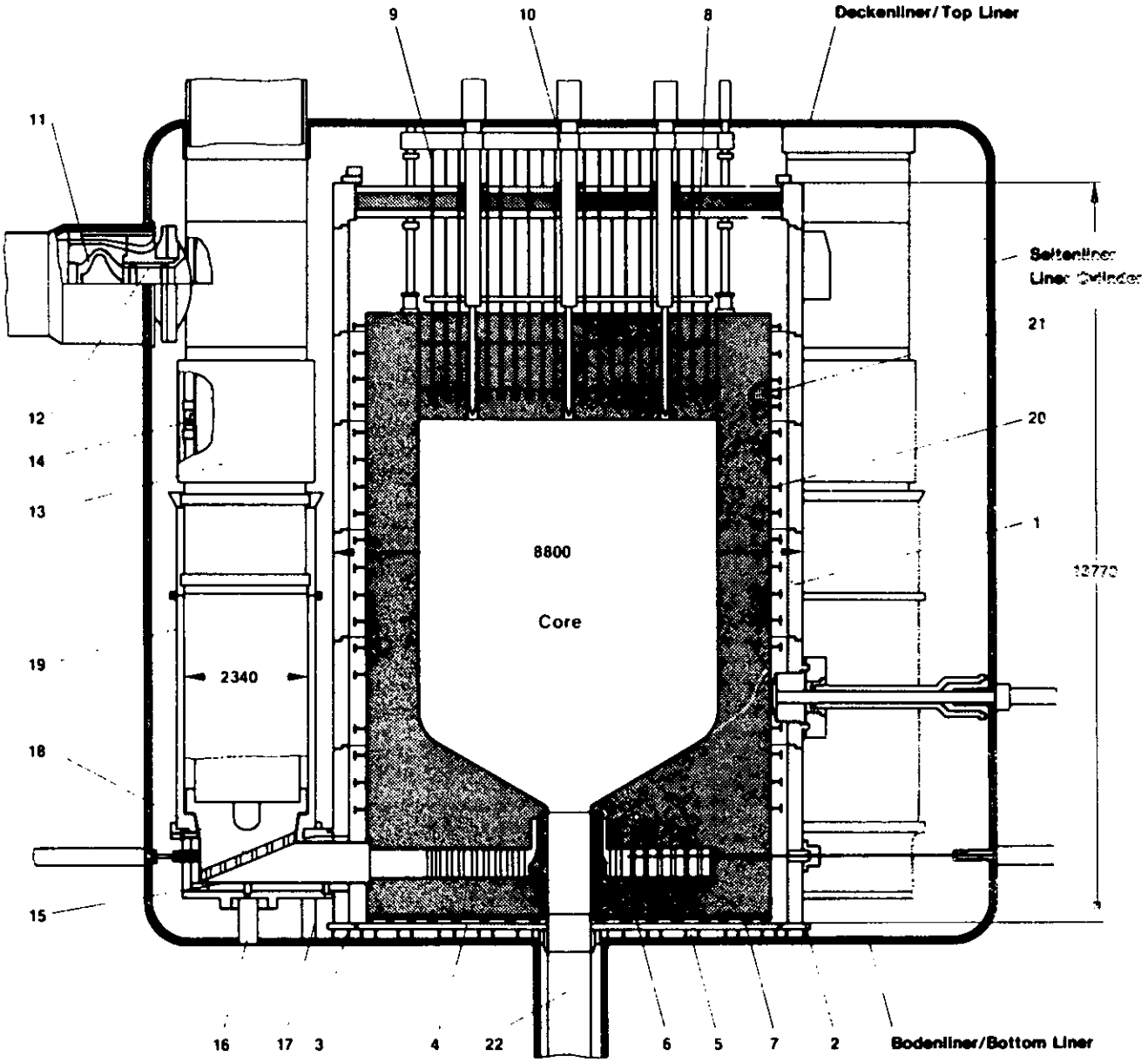


Abb. Die metallischen Einbauten des THTR-300

1 thermischer Seitenschild
2 Bodenplatte
3 Doppelrollenlager
4 untere Bodenplatte
5 Rollenlager
6 Bodenplatten
7 Doppelrollenlager
8 thermischer Deckenschild
9 Zugstangen
10 Tragringe
11 Gebläseabschirmungen
12 Saugleitungen
13 Dampferzeugermantel
14 Dichtelemente
15 Heißgaskanal
16 Fixpunkt
17 Doppelrollenlager
18 Lochplatte
19 Kaltgasfuhrungsmantel
20 Stutzbolzen
21 Verdrehsicherungen
22 Kugelabzugsrohr

Fig. 5 : THTR - 300 Metal Reactor Internals

1 Thermal Side Shield
2 Bottom Plate
3 Double Roller Bearing
4 Lower Bottom Plate
5 Roller Bearing
6 Upper Bottom Plates
7 Double Roller Bearing
8 Thermal Top Shield
9 Anchoring Rods
10 Support Rings
11 Circulator Shield
12 Suction Pipes
13 Steam Generator Metal Sheet Jackets
14 Sealing Elements
15 Hot Gas Duct
16 Fixed Point
17 Double Roller Bearing
18 Perforated Plate
19 Cold Gas Jackets
20 Spacer Bolts
21 Protection Devices against Torsion
22 Fuel Element Discharge Pipe

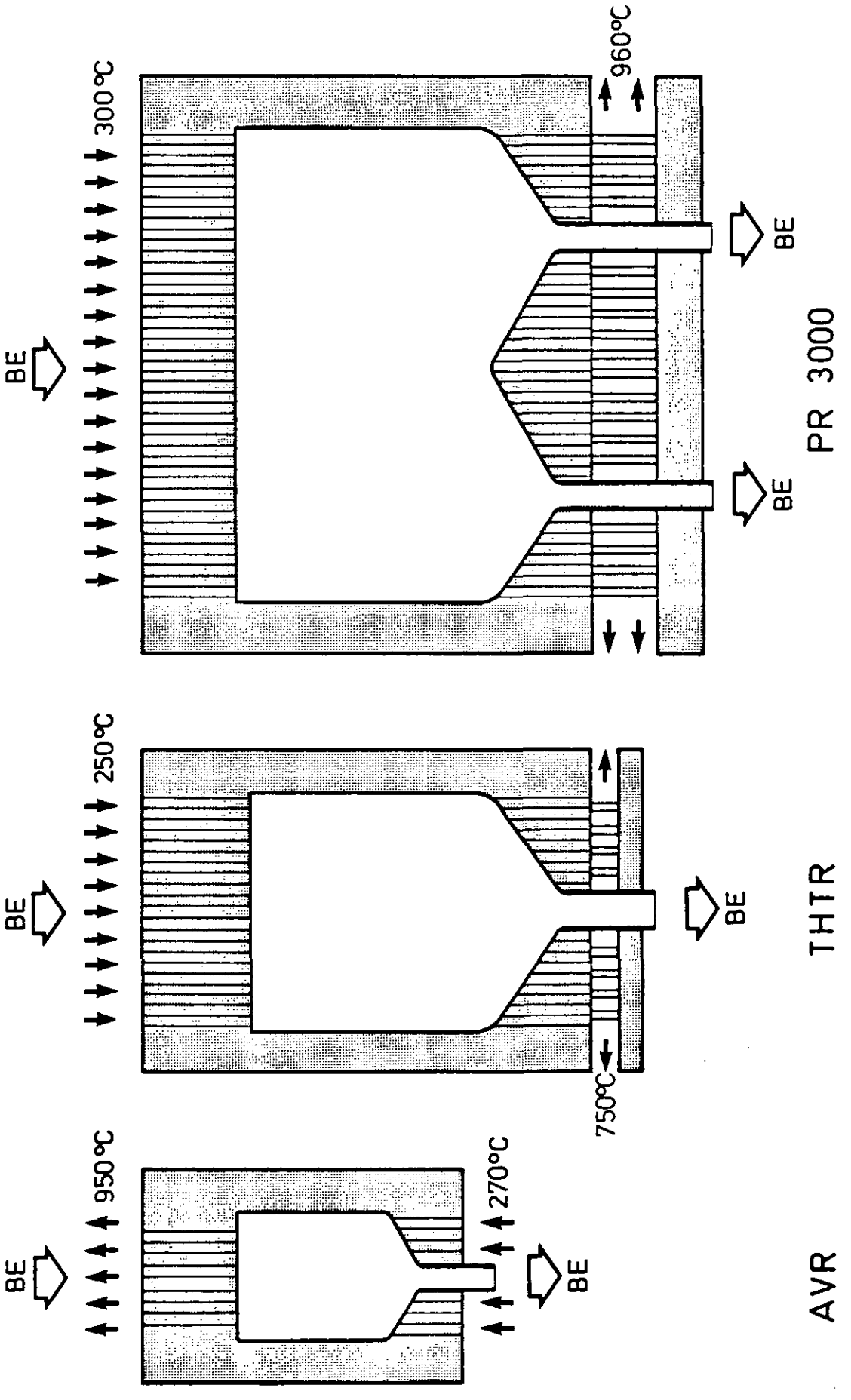

BE
300°C
960°C
BE
BE
PR 3000
BE
250°C
750°C
BE
THTR
BE
950°C
270°C
BE
AVR

Fig. 6

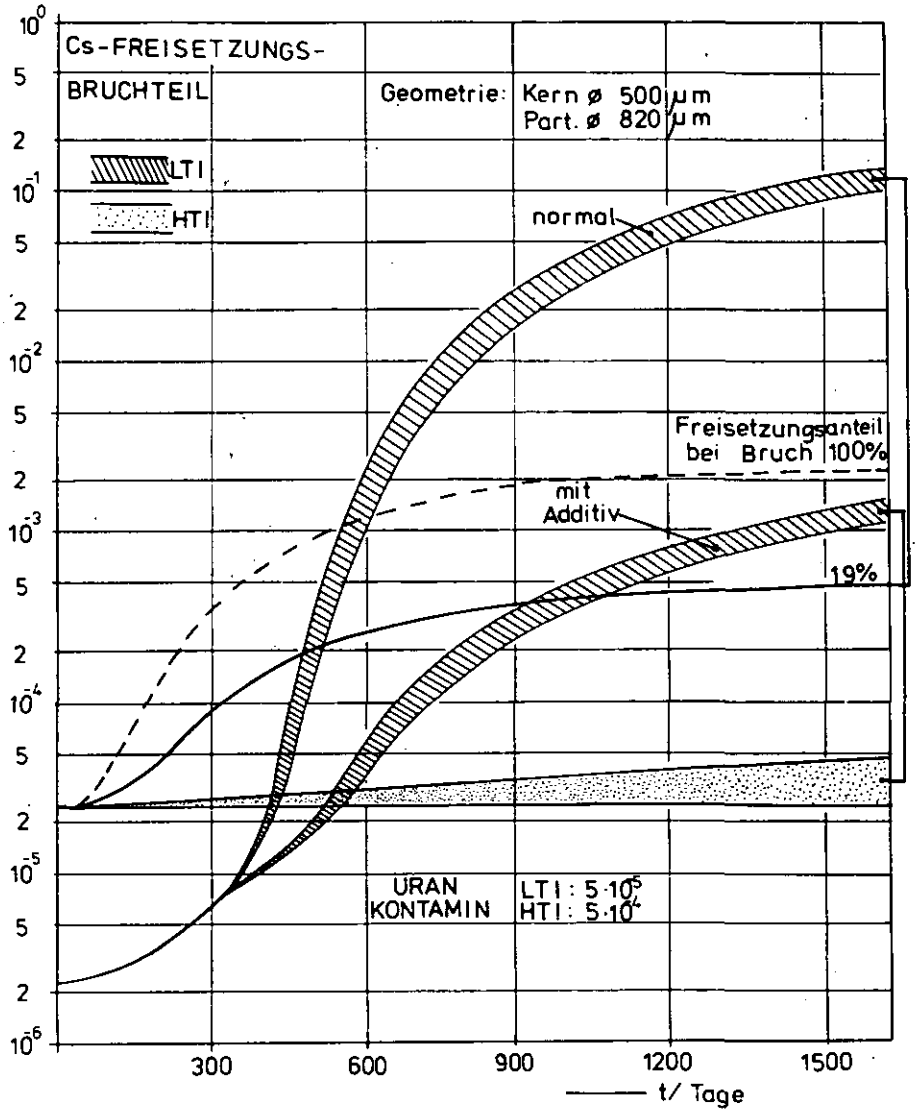
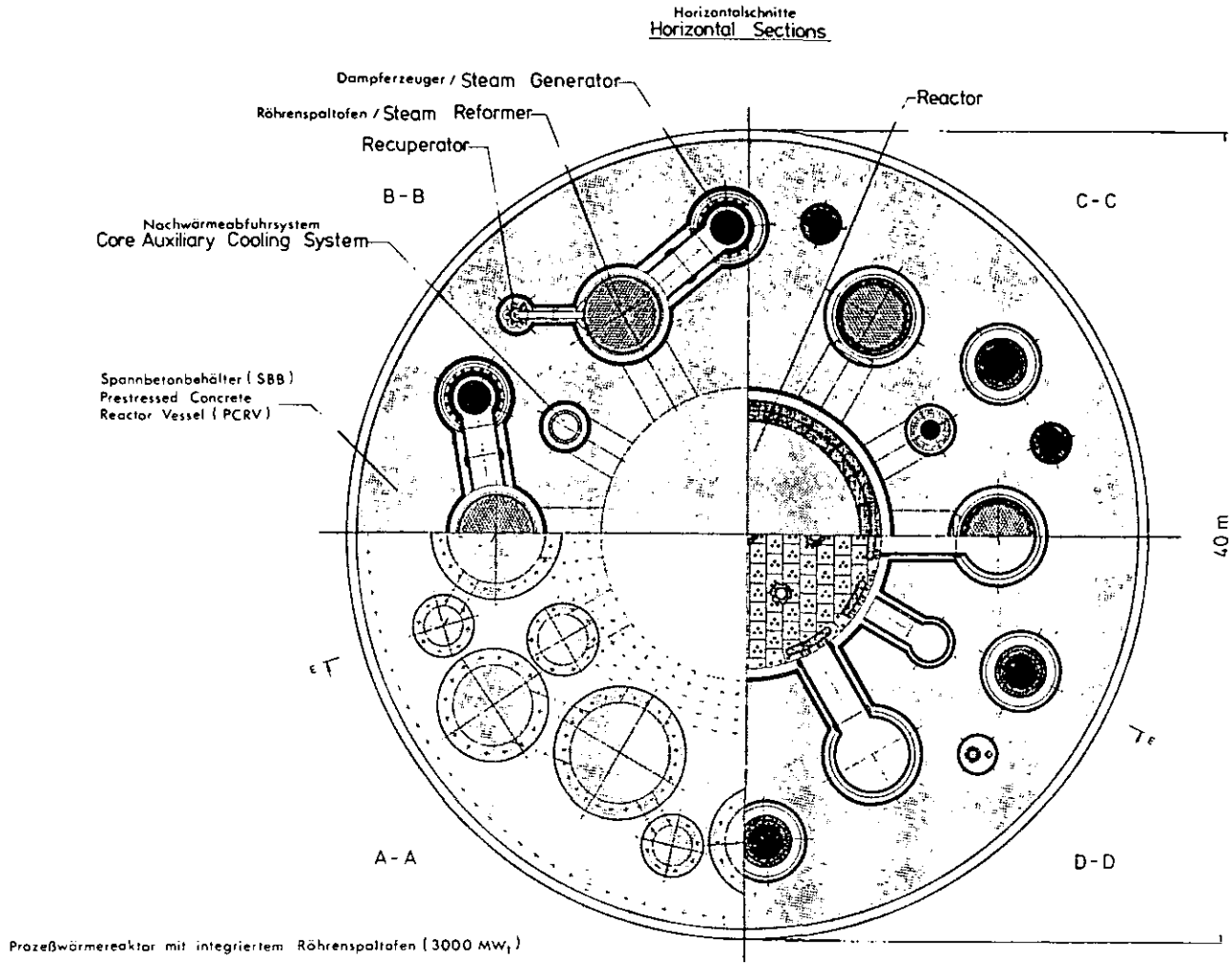
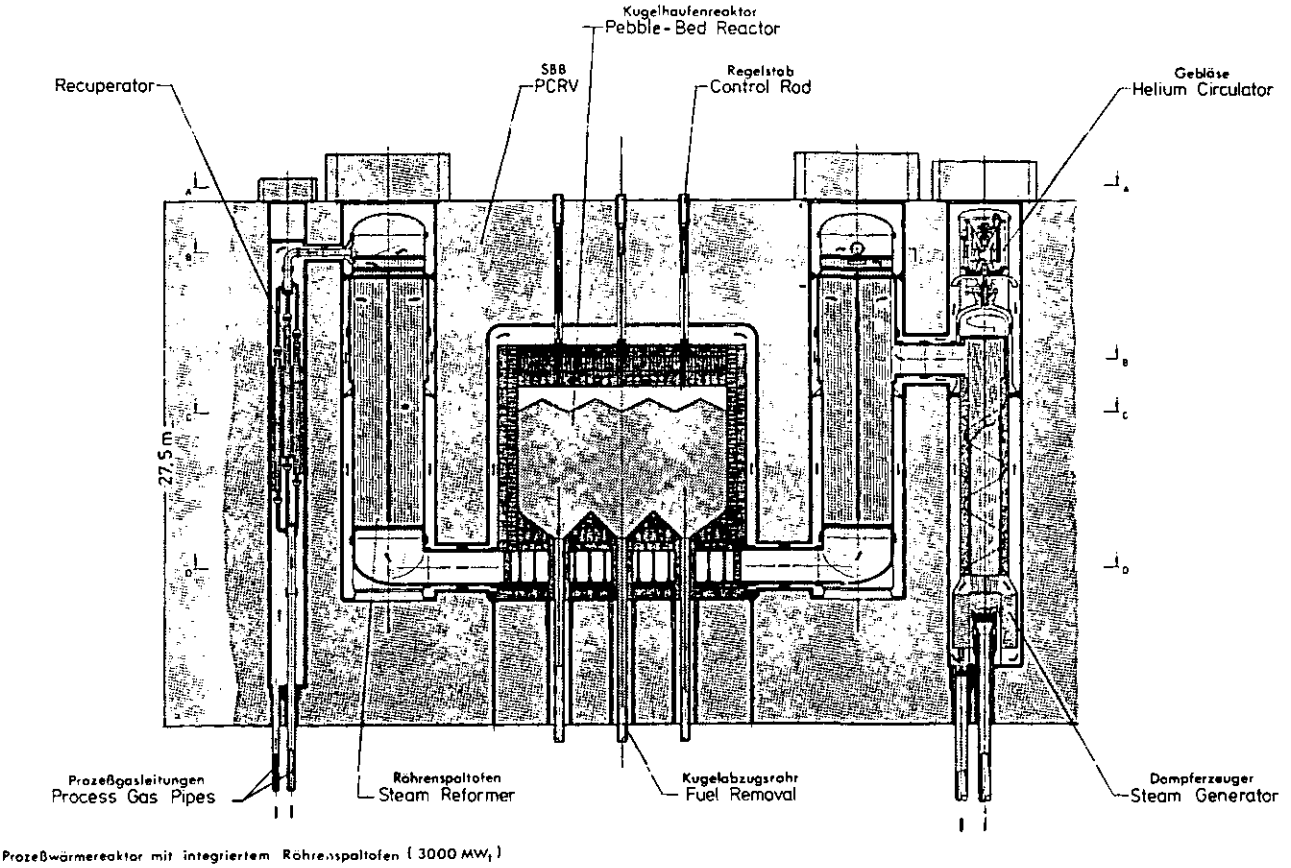


Abb.7: Cs-Freisetzung von BISO-Teilchen bei Durchlauf durch Zentralzone eines PNP-Core; Vergleich HTI/LTI



Prozeßwärmereaktor mit integriertem Röhrenspaltofen (3000 MW_t)

Fig. 8
Integrated 3000 MW reactor



Prozeßwärmereaktor mit integriertem Röhrenspaltofen (3000 MW_t)

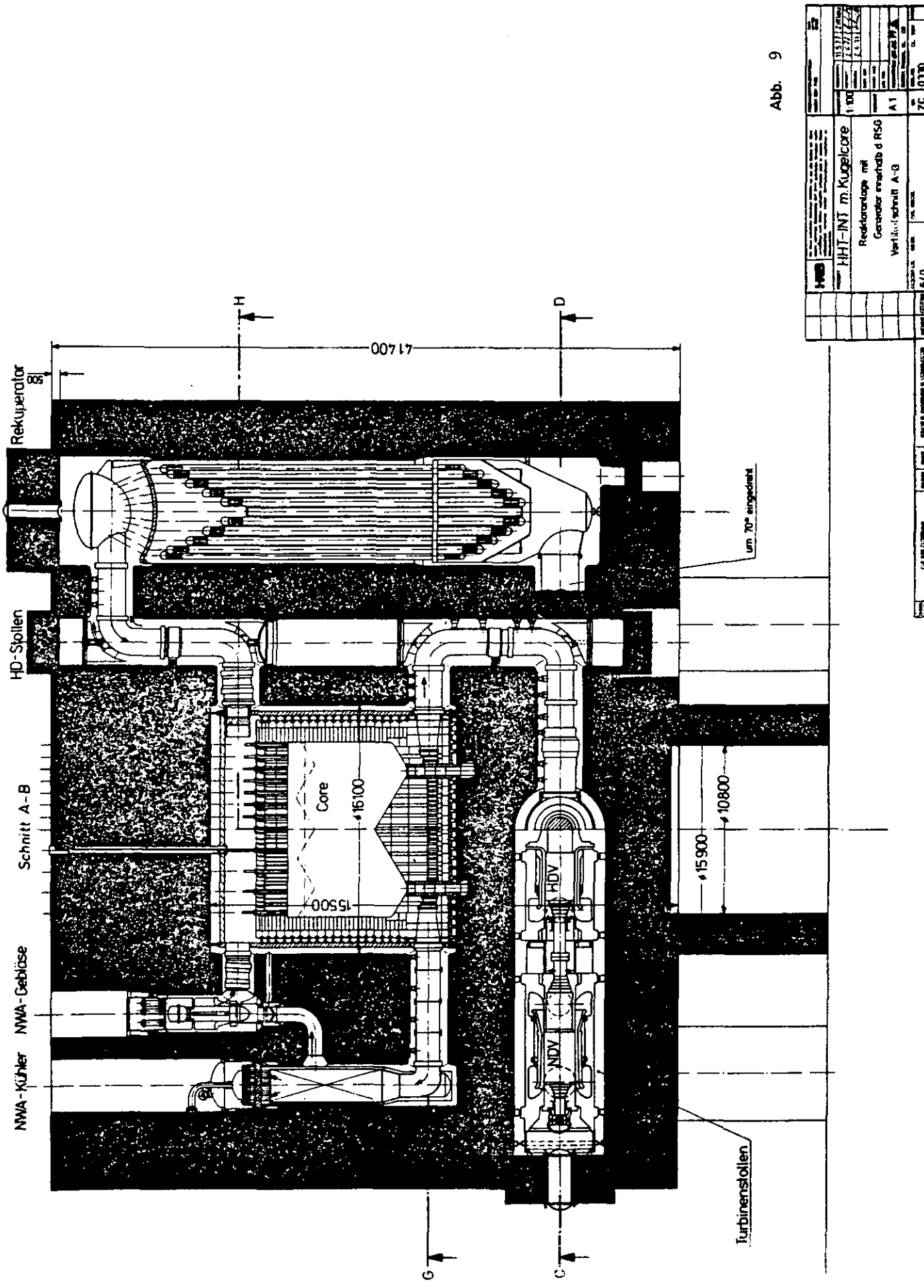

Rekuperator
HD-Stollen
Schnitt A-B
NWA-Gebläse
NWA-Kühler
H
D
G
C
41400
Core
16100
15500
HDV
NDV
um 70° eingedreht
15900
10800
Turbinenstollen
HHT-INT m.Kugelcore
Reaktoranlage mit
Generator maßstab d RSG
Vertikalschnitt A-B
A1
ZC 0330

Abb. 9

High Temperature Nuclear Process Heat of Coal Gasification and Long Distance Transport of Nuclear Energy

Paper presented by Dr. Hans Teggers, Rheinische Braunkohlenwerke AG, Köln at the HTR-Information-Seminar in Stockholm, January 11, 1978.

Introduction

The consumption of natural gas which meanwhile convers a high percentage of the total energy demand is expected to increase in the future. On the other hand the resources of natural gas cannot meet this demar on a long term basis. Therefore many companies and research institute are engaged in developing new techniques to gasify hard coal and brown coal to substitute natural gas (SNG). One technique is the substitution of process energy needed for gasification by sensible heat of high temperature nuclear reactors (HTR). The main part of secondary energy is consumed as domestic and industrial heat. A new system of long distance energy, also using sensible heat of high temperature nuclear reactors, will be referred too.

1. Fundamentals of coal gasification

The gasification of coal is an endothermic process. If one starts with the aim that the final product of gasification should be methane, which can be fed into the currently existing widespread network of natural gas pipelines, the following basic reaction-formula can be laid down:

$$2\ C + 2\ H_2O_{lq} = CH_4 + CO_2; \qquad (1)$$
$$\Delta H = + 79{,}1\ kJ$$

Corresponding endothermic gasification processes can be laid down in formulas for the production of hydrogen and carbonmonoxide containing mixtures or hydrogen.

$$C + H_2O_{lq} = CO + H_2; \qquad (2)$$
$$\Delta H = + 162{,}9\ kJ$$

$$C + 2\ H_2O_{lq} = CO_2 + 2\ H_2; \qquad (3)$$
$$\Delta H = + 165{,}4\ kJ$$

Figure 1: Basic gasification reactions.

The gasification of coal can in a simple way be shown as follows:

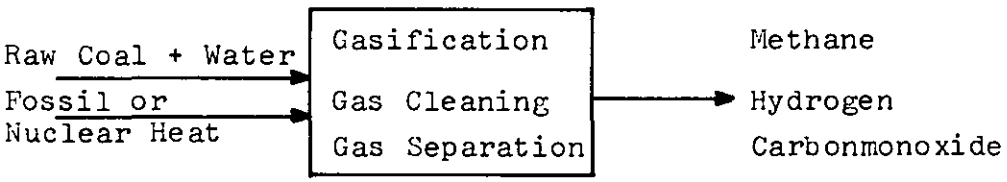


Figure 2: Black-box scheme of coal gasification.

Raw Coal, water and heat are introduced to the system, methane or alternatively a mixture of hydrogen and carbonmonoxide or pure hydrogen as well are the end products. Process heat is supplied either by burning part of the totally used coal or by substituting this process heat by the sensible heat of a high temperature gas cooled nuclear reactor.

2. Advantages of Coal gasification using nuclear heat

Replacing process heat from coal by nuclear process heat results in a decrease in gas production costs. Additionally, the application of nuclear process heat allows more coal to be converted to usable gas: by comparison with conventional gasification processes, about 30 - 40 % of the feed coal can be saved. Also, nuclear process heat contributes to the efforts of reducing environmental pollution as much less CO_2 is emitted than in conventional gasification processes. At the same time, the smaller rate of CO_2 production permits a decrease in the gas cleaning system, and thereby a decrease in specific plant costs. On the whole, the application of nuclear process heat is attractive because of a) the reduction in gas production costs, particularly in Europe, b) the savings of coal reserves; and c) the decrease of environmental problems.

3. Different ways of application nuclear heat to coal gasification

There are different ways of application nuclear heat to coal gasification. Two of these techniques which are developed in the framework of the project "Prototypanlage Nukleare Prozeßwärme" (PNP) undertaken by the companies Bergbau-Forschung GmbH, Gesellschaft für Hochtemperaturreaktortechnik mbH, Hochtemperatur-Reaktorbau GmbH, Kernforschungsanlage Jülich GmbH and Rheinische Braunkohlenwerke AG may be outlined. The project is sponsored by the Ministry of Research and Technology of the Federal Republic of Germany.

3.1 Hydrogasification

In the coal gasification with nuclear process heat Rheinische Braunkohlenwerke are developing the hydrogasification.
The basic chemical reactions for this process referred to pure carbon (the reaction enthalpies refer to char carbon) are (Figure 3):

Gasifier:

$$2\,C + 4H_2 = 2\,CH_4;\ \Delta H = -172{,}5\ kJ \qquad (1)$$

Methane steam reformer and shift conversion:

$$CH_4 + 2H_2O_g = CO_2 + 4\,H_2;\ \Delta H = +163{,}7\ kJ \qquad (2)$$

Steam generator:

$$2H_2O_{lq} = 2H_2O_g;\ \Delta H = +87{,}9\ kJ \qquad (3)$$

Total process:

$$2C + 2H_2O_{lq} = CH_4 + CO_2;\ \Delta H = +79{,}1\ kJ \qquad (4)$$

The coal is gasified with hydrogen in an exothermic reaction (1). Part of the produced methane is steam reformed and shift-converted to hydrogen and carbondioxide (2).
The heat for this processes including steam generation (3) is delivered from a high temperatur nuclear reactor. The net theoretical heat demand for the process is given by the summarizing reaction (4).

Most of the heat required for the hydrogasification process is coupled out from the nuclear loop by a Helium heated steam-reformer, in which methane is reformed with steam in an endothermic reaction into a mainly carbonmonoxide and hydrogen containing synthesis gas. In this process step on one hand the supply of process heat for the gasification is achieved, on the other hand the hydrogen balance of the gasification process is closed here.

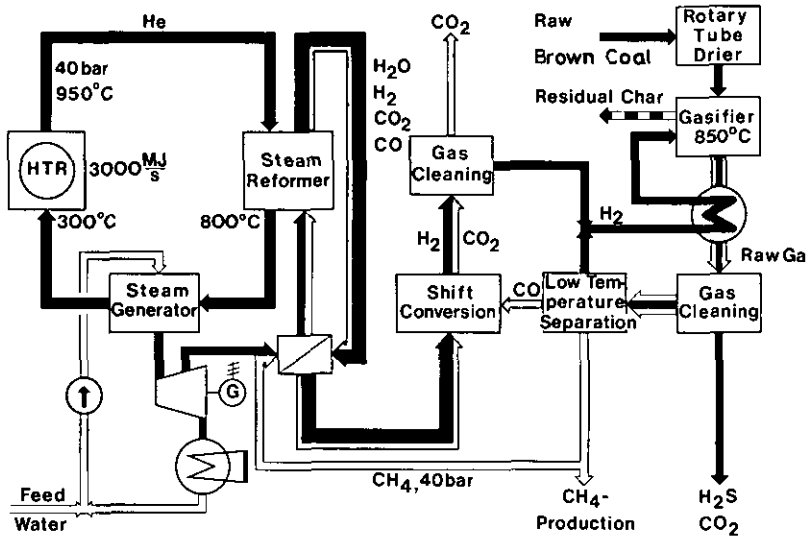


Figure 4: Hydrogasification of Brown-Coal

A general scheme of the hydrogasification process applied to brown coal as feed is shown in Figure 4.

Raw brown-coal is dried to a moisture content of 10 wt % and then fed into the gasifier in which it is gasified with hydrogen in a exothermic reaction in a fluidized bed at 850 °C and 80 bar.

Carbondioxide and hydrogen-sulfide in the raw-gas are removed. A low temperature separation splits the cleaned raw-gas into a methane-, a hydrogen-, and a carbonmonoxide-fraction. The hydrogen is recycled to the gasification reactor as gasifying agent. The larger part of the produced methane is compressed to 70 bar and disposed as SNG. The smaller part of the methane is fed into the steam-reformer where it is reformed in an endothermic reaction into carbonmonoxide and hydrogen at 800 °C. The required reaction heat is supplied by nuclear process heat.

The reformed gas is fed together with the carbonmonoxide from the low temperature gas separation unit into a shift conversion unit and finally as hydrogen into the gasifier.

3.2 Steam gasification

Within the above mentioned "PNP"-program another gasification process with nuclear heat is developed by the Bergbau-Forschung GmbH, Essen, namely the steam gasification of coal.
The basic equation of this process can be seen from Figure 1, equation (2). Figure 5 shows a general flow scheme of the process:

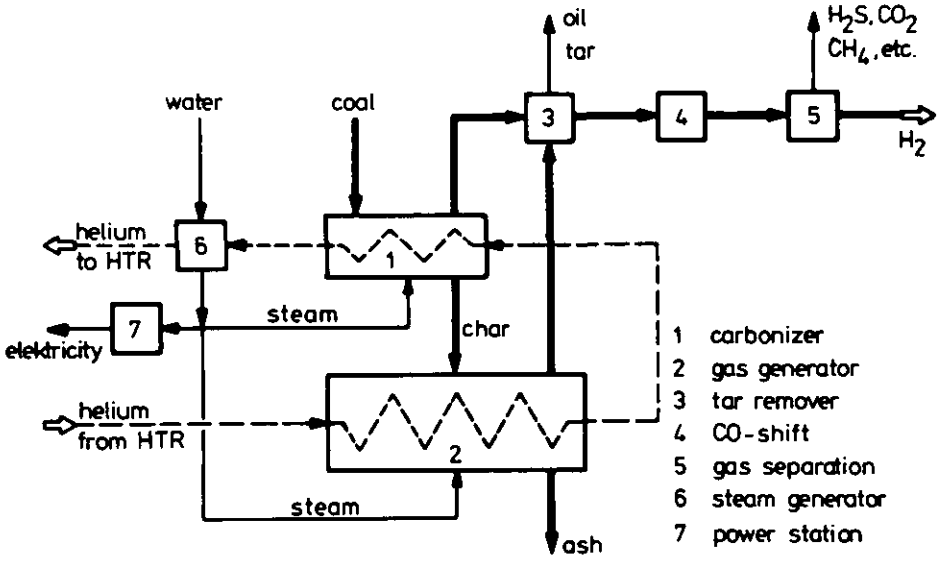


Figure 5: Hydrogen Production by Steam Gasification of Coal Using Nuclear Heat

The heat is coupled out from the primary Helium circuit of nuclear reactor by an intermediate heat exchanger to a secondary Helium circuit. This secondary Helium gas passes a heat exchanger immersed in the fluidized bed of coal using steam as fluidizing and gasifying agent. This heat exchanger provides the heat necessary for the gasification of the coal. It is advantageous that the primary helium loop, the secondary gas circuit and the fluidized bed operate at nearly the same pressure of about 40 bar. The steam required is generated and superheated with helium at a lower temperature level. The gas produced in the gas generator can be processed to synthesis gas or hydrogen or to SNG.

4. Development of components

Before erecting commercial scale units for coal gasification with nuclear process heat research and development work has to be done for several components. The actual status expecially for components on the gasification field may be described below.

4.1 The hydrogasifier

Hydrogasification is the conversion of coal carbon with hydrogen to methane according to the reaction

$$C + 2\ H_2 = CH_4; \quad \Delta H = -\ 86\ kJ$$

Hydrogasification takes place in two reaction steps with different velocities. In the first fast reaction step easily separable groups of coal containing hydrogen or oxygen are converted. In the second slower reaction step the reaction between hydrogen and the carbon skeleton, mainly consisting of condensed cyclic compounds, takes place.

Rheinische Braunkohlenwerke AG are testing hydrogasification in a semi-technical pilot plant under pressure in a fluidized bed. This plant was built on the basis of an information exchange with research institutes in the United States of America and Australia, and on the basis of many years operation experience in conventional gasification processes in fluidized beds. The plant has been designed to test the gasification on essential process parameters like pressure, temperature, residence time of gas and solid material, as well as type, pretreatment, and input of the feed coal. Furthermore, components of the

plant for which no operation experience was available until now were studied. To get informations on the behaviour of fluidized beds of dry brown coal or semi-char, respectively, preliminary experiments without gasifying the coal were performed in a tube with an inner diameter of 200 mm, which corresponded to the dimensions of the gasifier to be constructed. Nitrogen under normal conditions was used as fluidizing agent for these preliminary tests, which simulated the flow conditions of hydrogen at 80 bar and 800 °C.
The characteristic data of a fluidized bed as fluidizing velocity, elutriation velocity, density of the fluidized bed and its expansion depending on the velocity of the fluidizing agent were determined.

The experiments have shown that a homogeneous fluidized bed can be realized using dry brown coal with a particle size of 0 - 1 mm at gas velocities of only 2 - 3 cm/sec. Even up to gas velocities of more than 10 cm/sec and a bed height of 400 cm, the fluidized bed still shows a good homogeneity and only a small formation of bubbles.
A simplified flow scheme of a semi-technical pilot plant for the hydrogasification is shown in Figure 6.

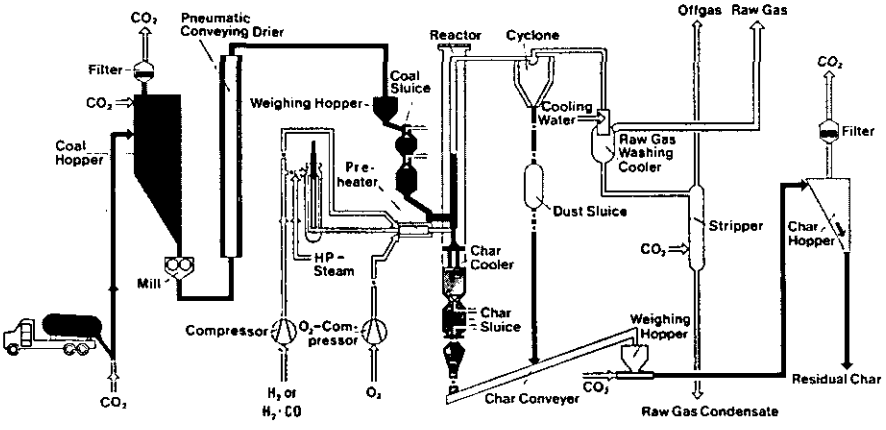


Figure 6: Semi-technical Pilot Plant for the Hydrogasification of Coal

It has a capacity of 200 kg/h dried brown-coal and is on operation since May 1975.

The delivered coal is stored, grinded, dried and fed to the gasifier by a lock hopper system. The gasifier itself is a fluidized bed reactor, designed for maximum pressure of 100 bar and temperatures up to 1 000 °C. The gasifier has an inner diameter of 200 mm and a height of 8 m; the height of the fluidized bed itself amounts up to about 4 m.
For the total conversion of coal to methane a rather long residence time of the solid material would be required caused by the second rather slow step of the hydrogasification. For economic reasons a total coal conversion is not envisaged; the finally received residual char is removed when a coal conversion of 70 to 80 % is achieved. The residual char is withdrawn, cooled down and depressurized in a lock hopper system which is similar to that in the feeding system. The residual char of the hydrogasification in general can be stored and used for further purposes, e. g. in combustion processes to generate electricity or for the nuclear steam gasification processes like the process of the Bergbau-Forschung described in this paper.

The raw-gas is mechanically cleaned by an internally insulated cyclone and cooled down by direct water scrubbing. A specially designed hydrogensulfide removal unit and a methane separation unit which would be used in a commercial plant is not installed in this plant. The raw-gas of the plant is disposed as a fuel gas.

An overall view of the semi-technical pilot plant is given in the Figure 7

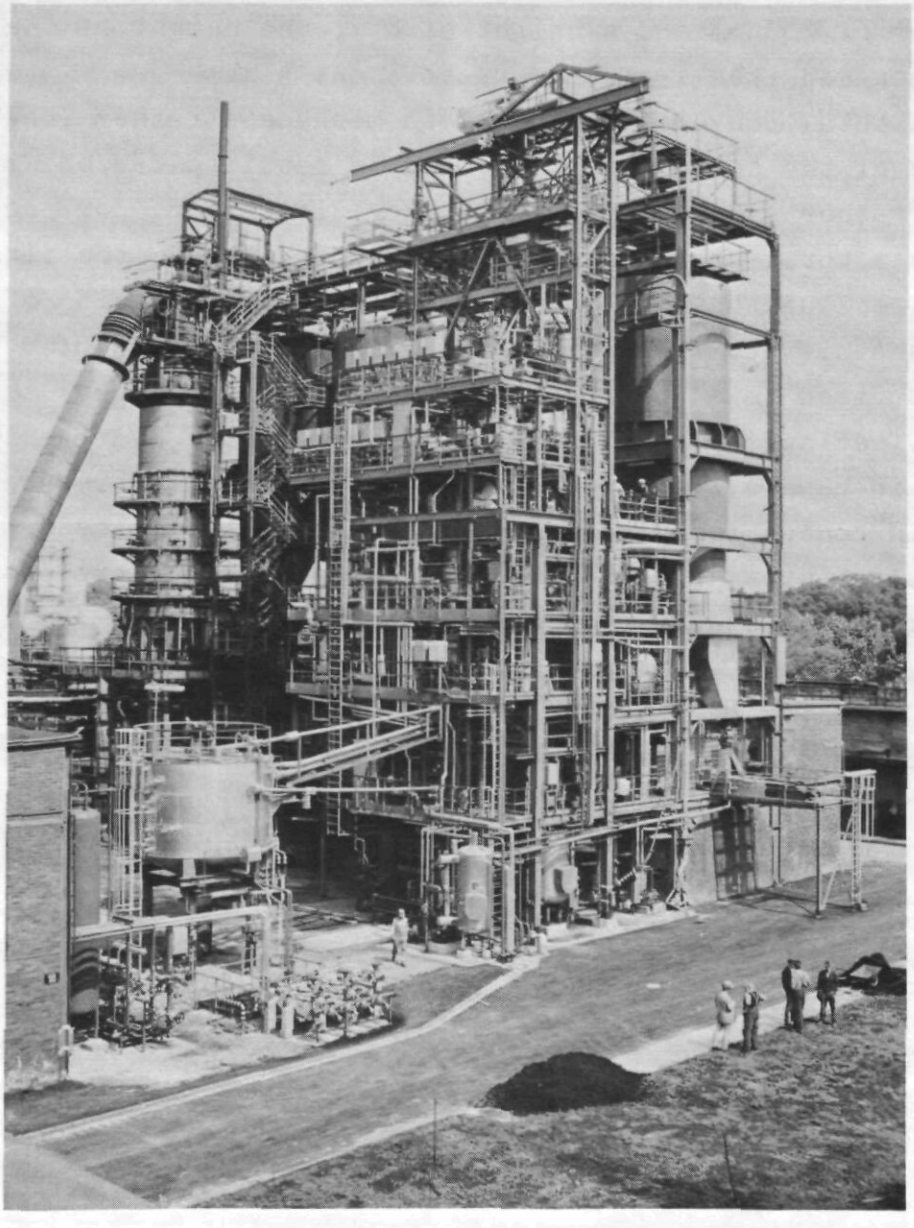


Figure 7: View of the Semi-technical Pilot Plant for Hydrogasification

An example of some important characteristical data during a test period in 1977 is shown in Figure 8

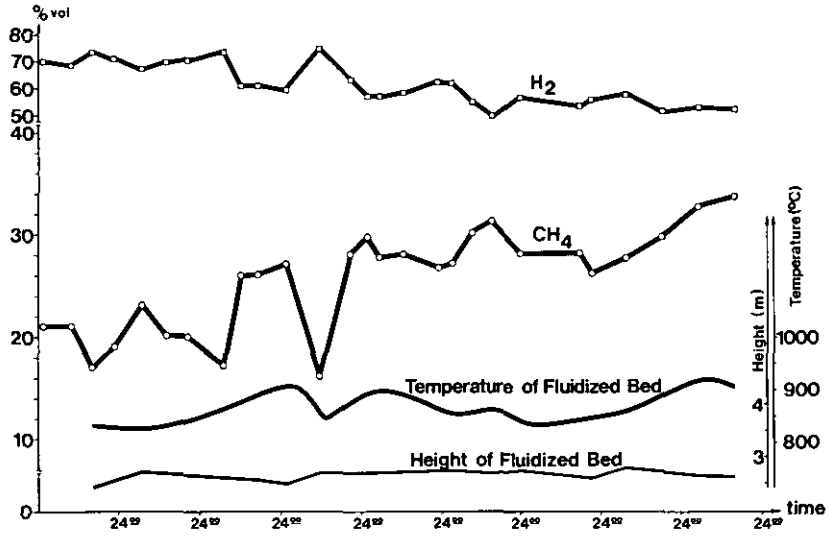


Figure 8: Hydrogasification of Brown Coal,
Example of a Test Run

In 1977 the plant was an operation nearly 5 000 hours for investigation of important process parameters of hydrogasification as well as for testing other problems concerning the process and apparatus, of which nearly 2 000 hours were with throughput of coal. During this time about 300 metric tons of dried brown coal were processed. The longest continuous period of operation lasted 367 hours with a total throughput of 55 metric tons of dried brown coal. In the course of the plant operation in 1977 the operation conditions were veryfied in large ranges:

The gasification pressure from 65 to 95 bar, the gasification temperature from 820 to 920 °C and the residence time of solid material from 15 to 65 minutes. A carbon gasification degree of max. 75 %, corresponding to a coal gasification degree of max. 80 %, a methane content in the raw gas of 48 vol. % and as well a specific production of more than 1 m^3 methane produced per kg of coal input was achieved. The throughput of coal could be raised up to 300 kg/h.
Essential data of hydrogasification of Rhenish brown coal in respect to relation of gasifier performance on coal throughput, temperature and pressure were obtained.

Also first preliminary tests on gasification of hard coal were performed.

Some test run data are given in Figure 9

Gasification Conditions

	Dimension	1	2	3
Gasification temperature	°C	890	865	920
Gasification pressure	bar	80	80	80
Height of the fluidized bed	m	2,7	2,7	2,6
Input of brown-coal (mf)	kg/h	265	300	212
Grain size	mm	< 1	< 1	< 1
Input of hydrogen	Nm^3/h	435	384	382

Results

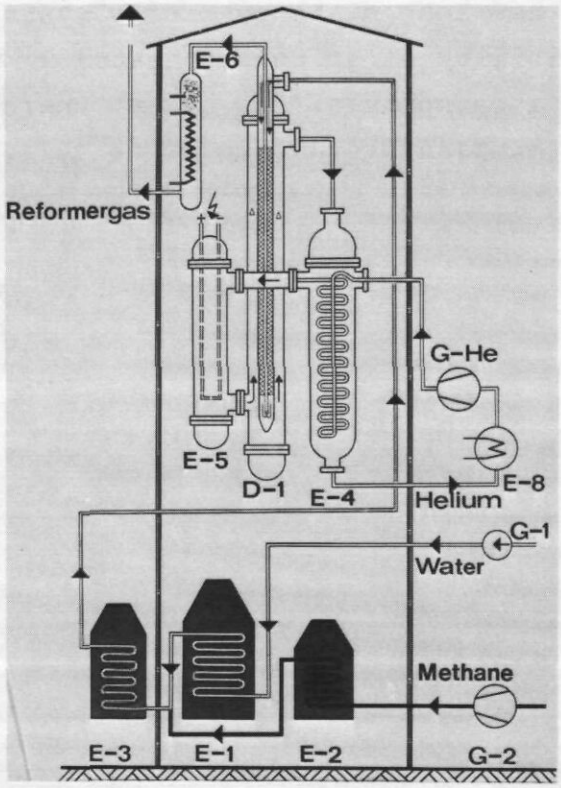
Carbon-Conversion		%	54,8	50,6	63,2
Composition of raw gas (dry, N_2 free)	CH_4	Vol-%	31,2	34,2	34,3
	C_2^+		2,3	3,4	2,5
	H_2		58,6	52,8	55,2
	CO_2		1,7	3,1	1,9
	CO		6,2	6,5	6,1
Yield of hydrocarbons			0,82	0,81	0,83
Residence time of char		min	~19	~16	~28

Figure 9: Test Run Data

4.2 The Methane-steam-reformer and his various applications

The methane-steam-reformer unit on the one hand is a main process element of the hydrogasification of coal. On the other hand this process element is main part of a new long distance energy system. For the demonstration of the feasibility of an helium heated steam-reformer a pilot plant with a capacity of 200 m^3 methane/h has been operated in the Kernforschungsanlage Jülich since 1972. (Figure 10 and Figure 11)

Figure 10: View of the Methane-Steam-Reformer Unit



D-1 Steam-Reformer
E-1 Steam-Boiler
E-2 Methane-Preheater
E-3 Steam-Methane-Superheater
E-4 Helium-Heat-Exchanger
E-5 Helium-Superheater
E-6 Reformergas-Cooler
E-8 Heat-Exchanger
G-1 Feedwater-Pump
G-2 Methane-Compressor
G-He Helium-Blower

Figure 11: Single-Tube-Reformer-Plant at KFA Jülich

Meanwhile Kernforschungsanlage Jülich GmbH and Rheinische Braunkohlenwerke AG started the projekt "Nukleare Fernenergie" (NFE), in which a pilot plant having 30 methane-steam-reformer tubes of a commercial size heated up by Helium will be erected. Construction will be finished in 1979.

The main characteristic of this nuclear based system of long distance energy is the combination of an endothermic chemical reaction taking place at the location of the high temperature nuclear heat source and the reverse exothermic chemical reaction on location of an area of energy and heat consumption. Or in other words: the energy of an high temperature nuclear reactor is transported as latent bound chemical energy.

A technically and economically applicable way of such a system is the combination of the endothermic reaction of the methane-steam *reforming* to a *mainly carbonmonoxide* and *hydrogen* containing reformer gas and the reverse exothermic methanation of this gas. The reformergas can be transported to a location in a distance of about 70 - 100 km where by methanation the energy and domestic or industrial heat needed are released. The fundamental equation of such a system is:

$$CH_4 + H_2O_{lq} = CO + 3H_2; \Delta H = +249{,}1 \text{ kJ}$$

Such a system can be conveniently accommodated into the existing infrastructure of town heating systems.

Figure 12 shows the basic flow scheme of this closed circuit.

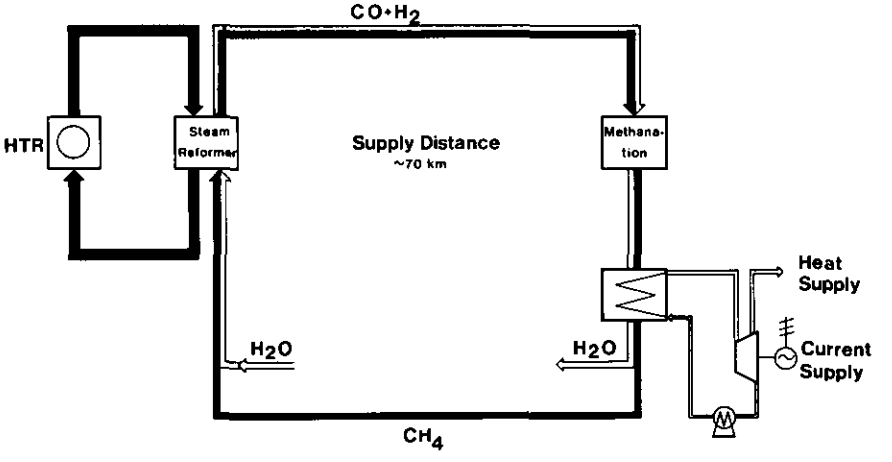


Figure 12: Nuclear Long-Distance Energy, Closed Circuit System

For a constant reactor power and helium outlet temperature, the nuclear thermal efficiency available for the steam-reforming plant depends considerably on the reactor inlet temperature of helium. At a reactor inlet temperatur of 350 °C in Figure 12 nearly 60 % of the reactor power can be converted into long-distance energy. If, in the future, it would be possible to attain an inlet temperature of 450 °C, then as much as about 75 % of the reactor power could be utilized. In this case, additional subquent aggregates would not be necessary except for the required steam generation for methane conversion.

In the concept shown in Figure 12, the remaining part of the reactor power can be used exclusiveley for electricity production or for a combination of electricity production and district heating.

This closed long-distance-energy system advanteageously can be transformed into an open circuit system as illustrated in Figure 13. In this case there is no need for a special methane-pipe, as methane can be fed into the natural gas network in the vicinity of the methanation plants. On the other hand, the methane necessary for the steam-reforming process could be taken from such a network, if one exists in the vicinity of the reactor site.

A combination of this system with coal gasification processes is also possible. Thus, energy as well as raw material could be transported by such pipe-line systems. The surplus methane in methanation plants could be supplied to the natural gas network. Furthermore, an interconnected operation of methanation stations with other synthesis gas consumers (e. g. methanol production or ore reduction plants) is possible. Such a system gives the possiblity for an integrated supply of substantial gaseous products and energy in densely populated areas on the basis of coal and nuclear energy.

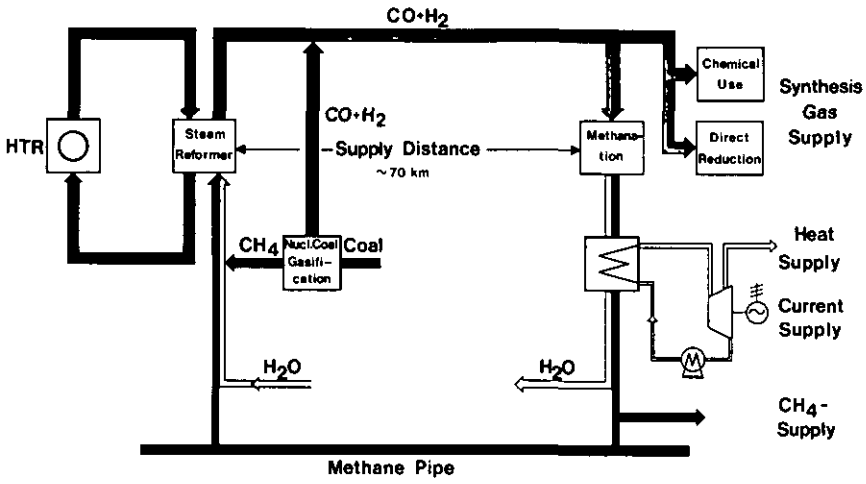


Figure 13: Nuclear Long-Distance Energy, Open Circuit System

4.3 The steam gasifier

The main investigation subjects of steam gasification with high temperature nuclear heat which are undertaken by Bergbau-Forschung in Essen are:

a) transfer of the sensible heat from the HTR into the gas generator
b) construction of an allothermal gas generator under the aspects of kinetics, heat transfer and materials for the immersed heat exchanger.

The design of such a commercial gasifier unit is shown in Figure 14.

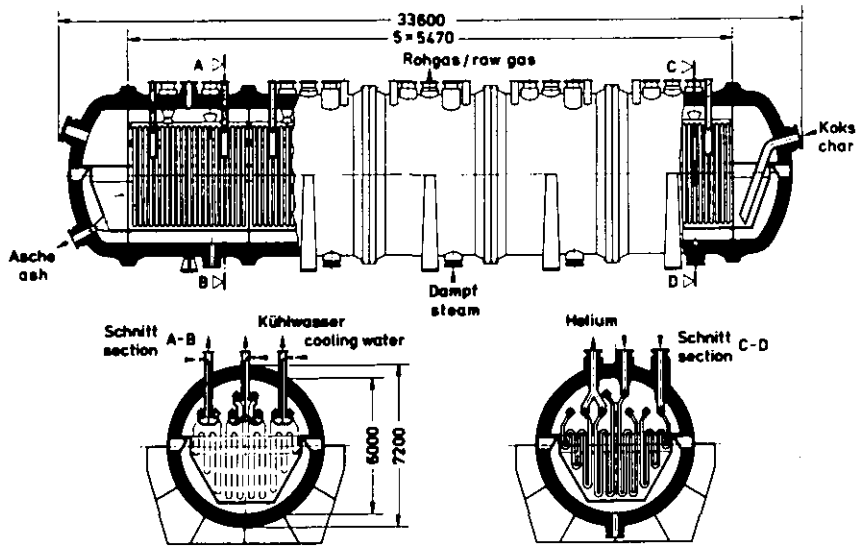


Figure 14: Design of an Industrial Gas Generator for the Steam Gasification of Coal.

The gasifier is a horizontal cylinder about 30 m long and 7 m in diameter. In this cylinder a special tube-heat exchanger system is immersed in a fluidized bed. This heat exchanger system is streamed through by the hot Helium of the secondary Helium loop. Coal is fed in through the inlet on the right-hand side on the top of the reactor. Ash can be withdrawn through the outlet on the left-hand side at the bottom. The bed is fluidized by high temperature steam injected from the bottom. An industrial gasifier for 50 t/h coal throughput needs a heat-exchanging area of say 4000 m^2; it has an effective volume for the fluidized bed of about 300 m^3. In co-operation with Mannesmann-röhren-Werke AG, Düsseldorf, the feasibility of this proposal has been proved and confirmed. One of the priority tasks is the development of a suitable alloy for the heat exchanger, which withstands corrosion, shows sufficient creep rupture strenght and allows tube forming. Experiments regarding these subjects meanwhile performed have shown promising results.

Main steps in the technical development of steam gasification in an allothermal gas generator up to now have been:

- Since 1969 the reaction kinetics of gasification with steam and hydrogen of different coals and chars have been investigated in a small fixed bed at temperatures up to 1000 °C, total pressures up to 70 bar, various partial pressures and using particle sizes smaller than 2 mm.

- Since 1973 a 5 kg/h test unit having an internally heated fluidized bed has been operated up to 40 bar. This unit has given the first results concerning reaction kinetics, gas composition and heat transfer under process conditions.

- In 1975 Bergbau-Forschung started to operate a semi-technical pilot plant, Figure 15, characterized mainly by two features:

  1. The reaction heat is supplied by a Helium-heat exchanger.
  2. The gasifier itself is a modulus of a commercial gasifier.

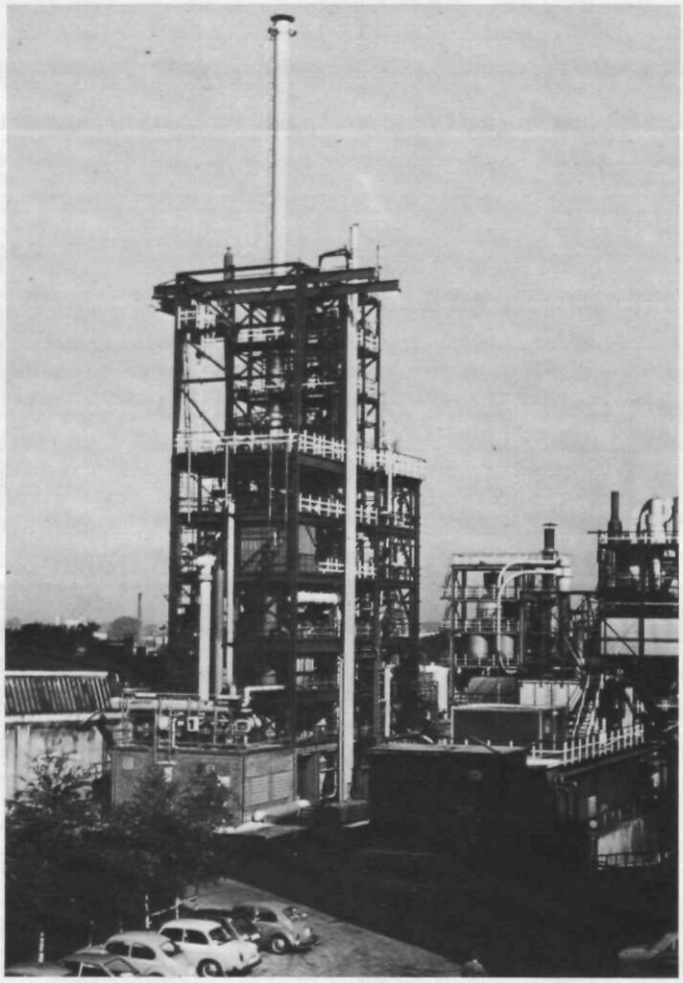


Figure 15: View of the Semi-technical Pilot Plant for Steam Gasification

5. The "Prototyp Nukleare Prozeßwärme" Project and it's time schedule

The semi-technical pilot plants of coal gasification and the methane-steam reforming unit are only intermediate steps on the way to large scale commercial plants. The semi-technical pilot plants, now having a coal throughput of 100 - 200 kg coal/h will be followed by pilot plants with a throughput of 5 - 10 t coal/h respectively a methane-steam-reformer with a troughput of 6000 m^3 CH_4/h. These pilot plants are planned to be on operation in 1980/81. Up to 1984/85 the detail engineering of a so called Prototyp Plant shall be finished, in which a high temperature nuclear reactor of a performance of about 500 MW thermal is connected with gasifiers for the steam gasification and

the hydrogasification as well as with methane-steam reformers. The single gasifier in this prototyp plant will already have a unit throughput which is nearly the same as in a commercial gasification plant, where the HTR will have a size of 3000 MW thermal.

6. Estimating the costs of the produced SNG

In 1976 a study on the market chances and the costs of SNG produced by coal gasification with nuclear heat was made.

From the calculations resulted that the costs of SNG at the time of startup of large scale commercial plant, i. e. in 1995 to 2 000 can compete with natural gas resp. domestic fuel oil under assumptions which seem reasonable. Figure 16 gives a survey on the results of these calculations.

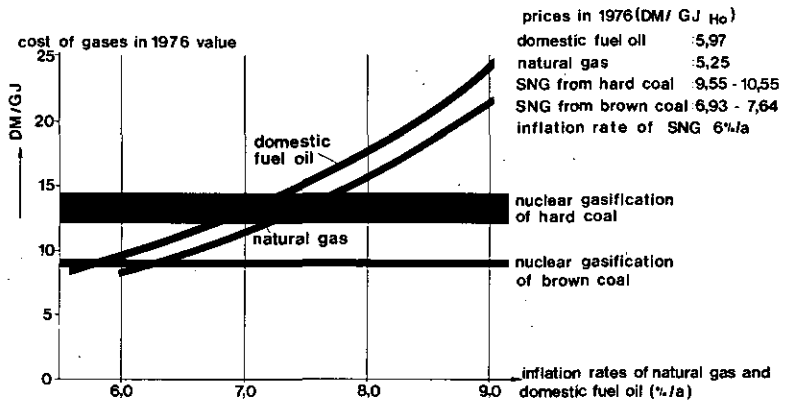


Figure 16: Cost of gases (20 years average) of commercial nuclear gasification plants going on stream in the year 2000 compared with natural gas and domestic fuel oil

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HTR FUEL CYCLES

by

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KFA Jülich GmbH

1. Reactor Physics Considerations

Small neutron losses in the graphite moderator and the absence of neutron absorbing structural materials in the core of the High Temperature Reactor (HTR) lead to a good neutron economy with a theoretical potential of a net fissile breeding gain. Due to various technical and economic constraints the present designs have not aimed at utilizing this potential to the full but rather to couple a resonably low uranium consumption with favourable economics.

The neutron moderation takes place in graphite and the heat is removed from the core by helium gas. The coolant does not absorb or slow down neutrons so that thermodynamic considerations do not impair the reactor physics.

The coated particles, in themselves fully sealed fuel elements are dispersed in the graphite fuel body. Changing the coated particle fraction in the graphite is in effect a way of varying the moderation ratio in the core. The high heavy metal loading needed for a near-breeder core for instance is achieved without altering the geometric dimensions of the fuel element. Furthermore, the coated particle can be designed to suit a wide variety of burnup and temperature conditions.

All this add to the possibilities of the designer to optimize the core for a given fuel cycle with negligible change to the standard HTR design und with minimal effect on the safety characteristics.

The two present fuel element designs, the block and the pebble, are very similiar in reactors physics respects. They contain the same types of c.p. and they may be designed for a similiar range of moderation ratios and burnups.

The pebble bed offers additional advantages in the fuel management operation,

- continuous refuelling and therefore no burnable poison for long term reactivity control
- simple one pass ball flow (OTTO cycle) /1,2/
- separation of feed and breed fuel in different fuel elements and so simpler reprocessing and refabrication.

The HTR may operate either on U/Pu, Th/U or a combination of both. The main work of development has concentrated on the thorium/U 233 cycle, although extensive core studies and fuel testing have been performed for the low enriched uranium cycle as well. The neutron characteristics of the bred materials show that Pu 239 and Pu 241 do better in a fast neutron spectrum whereas U 233 is superior in a thermal one.

Reprocessing of thorium/uranium fuel is not yet available, although work on the THOREX process is well in hand. The closing of the thorium fuel cycle is essential to exploit fully the high fuel utilization in the HTR.

The following discussion on once-through and closed fuel cycles is based on the standard 3000 MW(th) pebble bed core (Tab. 1).

2. Once-Through Fuel Cycles

The first HTRs will operate on an open-ended fuel cycle similiar to present light water reactors. Consequently, the HTR fuel cycle will be optimized towards once-through operation without credit for discharged fissile material. This does not prejudice reprocessing at some later time, but the optimisation reflects the long lag time with a correspondingly heavy financing burden.

The fuel cycles defined in Table 2 are all based on mixed oxide coated particles. Feed/breed concepts are feasible and may have advantages when a future reprocessing is considered. In such a case the discharged breed elements would contain the bulk of the residual fissile material and only these would be worth while to reprocess. /3,4/ The fuel for Th-HEU (High-Enriched Uranium) cycle uses the same coated particle as the THTR 300 MWe prototype. The heavy metal loading per ball is within the range of the AVR and THTR reference designs. The coated particle for the LEU (Low Enriched Uranium) cycle is similiar to the fuel developed at Dragon and KFA. From the broad field of development work and irradiation tests with fuels of various enrichment degrees and thorium contents the fuel for the Th-MEU (Medium Enriched Uranium) cycle has been selected. Specific demonstration test will have to be performed.

The HTR once-through cycles are compared for the same basic core layout and a discharge burnup of 100.000 MWd/t. As a point of reference the data for a large PWR of KWU standard design has been included in table 3.

The specific fissile inventory per net electric output is smaller in the HTR and amounts to approximately half of that in a PWR.

A similiar characteristic feature is that the discharged fissile material is of the same comparably low percentage. The gross natural uranium demand, i.e. without credit for recoverable fissile isotopes in the spent fuel, accumulated over a 25 years lifetime is

64 % for the HTR-Th/HEU cycle,
69 % for the HTR-Th/MEU cycle,
74 % for the HTR-LEU cycle,

all figures compared to the standard PWR.

Of the bred plutonium in the uranium cycles 90 % is burnt in-situ in the HTR and 80 % in the PWR. The in-situ utilization of U 233 is only 75 % and this is due to the smaller absorption cross section in this nuclide.

It has for political reasons been suggested to operate nuclear reactors on thorium and denatured uranium with 20 % enrichment (MEU). In this way the quality and quantity of fissile plutonium in spent fuel can be reduced to a level which is assumed insensitive with respect to proliferation of material suitable for atomic weapons. In the presentation above it has been shown that the HTR can operate succesfully with this fuel. The discharged fissile plutonium is reduced to roughly 20 kg/GWa(e) or less than a tenth of that contained in spent fuel from present LWRs. The isotopes Pu 239 and Pu 241 represent some 35 % of the total plutonium in spent HTR fuel.

For once through cycles it can be concluded that the HTR would decrease the demand on natural uranium by 25 to 35 % compared to present LWRs. The Th/HEU cycle is most effective in this respect (Fig. 1).

3. Closed Fuel Cycle

The recovery and recycle of fissile material in spent fuel improves the fuel utilization. Due to the physics properties of U 233 and Pu 239 the closing of the fuel cycle is of greater importance for thorium than for uranium in thermal reactors. /4,5/

The work on HTR fuel cycles has right from the beginning been aimed at exploting the thorium/U 233 chain.

Obviously, an increase in the conversion of Th 232 to U 233 will decrease the demand for natural uranium. But the recycling of bred fissile material must be discussed in the context of practical considerations such as

- the frequency of recycling and the fraction of refabricated fuel elements
- the control of the build-up of the parasitic absorber U 236.

The recycle strategies may deploy different fuel element types in the refuelling charges and the reprocessing handles either one common fuel stream or a separation of the various fuel types into two or more streams. In pebble bed reactors the separation takes place on the fuel element level and in block type reactors a segregation scheme with different coated particles is prefered. The aim of separation is to isolate the residual fuel from the fresh feed stream with its high U 236 content. This fuel may either be disposed of after one irradiation period or refabricated into special fuel for additional irradiation. The rejected U 236 will always be accompanied by a certain amount of fissile U 235. It is not clear if the increase in the fissile inventory to compensate for the accumulation of U 236 in the

core with "eternal" recycling is more costly than the disposal of U 235 in the segregated cycles. In any case the separation at the reprocessing stage is more complicated. A similiar problem will arise with LWRs and the uranium cycle. Blending of the recovered uranium with fresh material will lead to a recycle of all U 236 and re-enrichment will separate away only 40 % of the U 236 to the tails. /6/.

In table 4 the data is given for a fuel cycle similiar to the segregated concept proposed by the GAC for their original 1160 MWe stations. Designated HRS, this cycle in the pebble bed reactor has been designed around the GAC type of c.p. and the THTR fuel element and achieves only a moderately high conversion ratio. A lower burnup and higher thorium loading with recycling of all discharge uranium including the isotope 236 in mixed oxide particles characterise the second case, HRM. The fuel cycle is laid out with a conversion ratio of 0.74. A conversion ratio of 0.95 and even higher is achieved with a further increase in thorium loading and shortened burnup. This, however, requires U 233 for the start-up core and as make-up in subsequent reloadings. In a hard neutron spectrum U 235 is much inferior to U 233. The initial system inventory of U 233 may come either from normal HTR converters or specially designed pre-breeders with separation of feed and breed fuel streams in the reprocessing. /7/.

In discussing the results of fuel economy it is important to distinguish between the requirements for uranium and separative work for a single reactor during its period of operation and a system of such reactors. The fissile inventory of the last core and the out-of-pile stream will become evailable only after the reactor has been shut down and will not benefit the operator of that reactor during its lifetime. For the system, however, this

is an import asset for which a credit must be given. In particular for high converting systems with large inventories this credit can amount to a significant portion.

The results given in table 5 and fig. 2 show that with a good converter like the HRM the gross consumption of natural uranium during the lifetime of one reactor can be lowered to 58 % of that of a PWR with U+Pu recycle. Accounting for the fissile inventories of the two systems the net consumption figure is improved to 40 %.

The introduction of HTR near breeder cycles require the availability of U 233 and for this purpose pre-breeders are operated. In these half of the fuel in a reload batch contain ThO_2 breed elements which are reprocessed separately and the recovered U 233 used for nearbreeders. The gross uranium consumption of a PB is some 90% of that of a PWR on the oncethrough cycle.However, the U 233 output over a lifetime is sufficient to supply the net consumption of a system of 8.4 NB and hence to reduce the demand

	Nat. Uranium /tU/GWa(e)/	Sep. Work /tSWU/GWa(e)/
PB gross consumption	202	215
PB net consumption	180	192
1 PB + 8.4 NB net consumption in equilibrium	19	20

Table 6: Fuel Utilization in High Converting HTRs

on the natural resource uranium by a factor of 7 compared to a LWR system with Pu+U recycling (Table 6). But it must be stressed that these favourable results apply only to a large system of PBs and NBs in a zero growth energy economy.

During the introductional phase of the PB/NB systems the uranium demand is governed by the number of PBs and the time needed to build up the U 233 inventory of NBs. The PB suggested here would have to operate 12 to 14 years to accumulate sufficient U 233 to start up a NB and supply the reloads for the first three years. Thereafter the NB is virtually selfsufficient in fuel. When the fist NB is closed down the system inventory becomes available for the start-up of the next in line of NBs. The effective uranium ore savings will depend very much on the reactor strategy and the growth rate of the energy system.

4. Conclusions

The HTR can operate on a wide range of fuel cycles with only negligible changes to the basic reactor concept. All cycles presented here have been laid-out to comply with control requirements and engineering constraints on fuel and core structural components. In particular a temperature margin of 200 ^{o}C below the assumed fuel temperature limit of 1250 ^{o}C has been observed.

In view of conserving natural energy resources the main findings are

- In the once-through operated mode the HTR will decrease the natural uranium demand by 25 to 35 % compared to PWRs.

- The Th/HEU cycle is most favourable and could with recycling lower the net uranium consumption to 40 % of that of a PWR with U+Pu recycle.

- In the long term significant benefits may accrue from high converting cycles like the PB/NB system, in particular in a slowly expanding or zero-growth energy economy.

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MAIN REACTOR DATA		
THERMAL POWER	3000	MW
NET EFFICIENCY	41,4	%
POWER DENSITY	5	MW/M^3
CORE INLET TEMPERATURE	250	^oC
CORE OUTLET TEMPERATURE	985	^oC
MAX FUEL TEMPERATURE	1000 - 1250	^oC
CORE DIAMETER	11,8	M
CORE HEIGHT (TOP OF BALL LAYER)	5,5	M
FUEL BALL DIAMETER	6,0	CM
THICKNESS OF FUEL FREE ZONE	0,5	CM
ONCE THROUGH THEN OUT (OTTO) FUEL MANAGEMENT		

TABLE 1

ONCE-THROUGH FUEL FOR HTR	
HO:	HIGH-ENRICHED URANIUM (93 %)/THORIUM MIXED OXIDE IN THTR COATED PARTICLES HEAVY METAL LOADING 11,2 G/BALL
LO:	LOW-ENRICHED URANIUM (7-10 %) URANIUM OXIDE IN DRAGON TYPE COATED PARTICLES HEAVY METAL LOADING 9,9 - 11,7 G/BALL
MO:	MEDIUM-ENRICHED URANIUM (20 %)/THORIUM MIXED OXIDE IN THTR TYPE COATED PARTICLES HEAVY METAL LOADING 8,1 G/BALL

TABLE 2

ONCE-THROUGH FUEL CYCLE PARAMETERS					
		HO	MO	LO	PWR
MODERATION RATIO	C/HM	325	458	366	-
AVERAGE FEED FISSILE/HM	%	7,2	7,8	8,6	3,1
AVERAGE BURNUP	MWD/KG	100	100	100	32
CONVERSION RATIO	-	0,594	0,575	0,575	0,62
FISSILE INVENTORY	KG/GW(E)	886	683	988	1610
RELOAD U 235	KG/GWA(E)	638	688	741	1079
DISCHARGE U 233+ U 235	KG/GWA(E)	194	167	120	278
DISCHARGE PU 239+PU 241	KG/GWA(E)	1	19	81	223

TABLE 3

CLOSED FUEL CYCLES FOR HTR	
HRS:	HIGH ENRICHED URANIUM (93 %)/THORIUM. RECOVERED U FROM BREED ELEMENT RECYCLED ONCE IN FEED ELEMENT. HM LOADING 15 G(U+TH)/BALL AND 1,7 G(U)/BALL.
HRM:	HIGH ENRICHED URANIUM (93 %) MAKE-UP. ALL RECOVERED U RECYCLED WITH TH. HM LOADING 22 G(U+TH)/BALL, FRESH AND RECYCLED FUEL.
PB:	HIGH-ENRICHED URANIUM (93 %) MAKE-UP. RECOVERED U FROM TH-BREED ELEMENT RESERVED FOR NB. HM LOADING 5,2 G (U+TH)/BALL, 1,8 G (U)/BALL AND 32 G (TH)/BALL.
NB:	URANIUM 233 INITIAL CORE AND MAKE-UP. ALL RECOVERD U RECYCLED WITH TH. HM LOADING 32 G (U+TH)/BALL AND 32 G(TH)/BALL

TABLE 4

CLOSED FUEL CYCLE PARAMETERS					
		HRS	HRM	PB	NB
MODERATION RATIO	C/HM	325	160	198	110
AVERAGE FEED FISSILE/HM	%	6,9	5,7	3,4	3,4
AVERAGE BURNUP	MWD/KG	100	69	23	24
CONVERSION RATIO	-	0,62	0,74	0,74	0,97
FISSILE INVENTORY	KG/GW(E)	905	2200	1330	2800
FISSILE RELOAD	KG/GWA(E)	610	640	1309	1281
DISCHARGE U 233+ U 235	KG/GWA(E)	213	380	1015	1249
REMOVED U 233+ U 235	KG/GWA(E)	15	-	485	-

TABLE 5

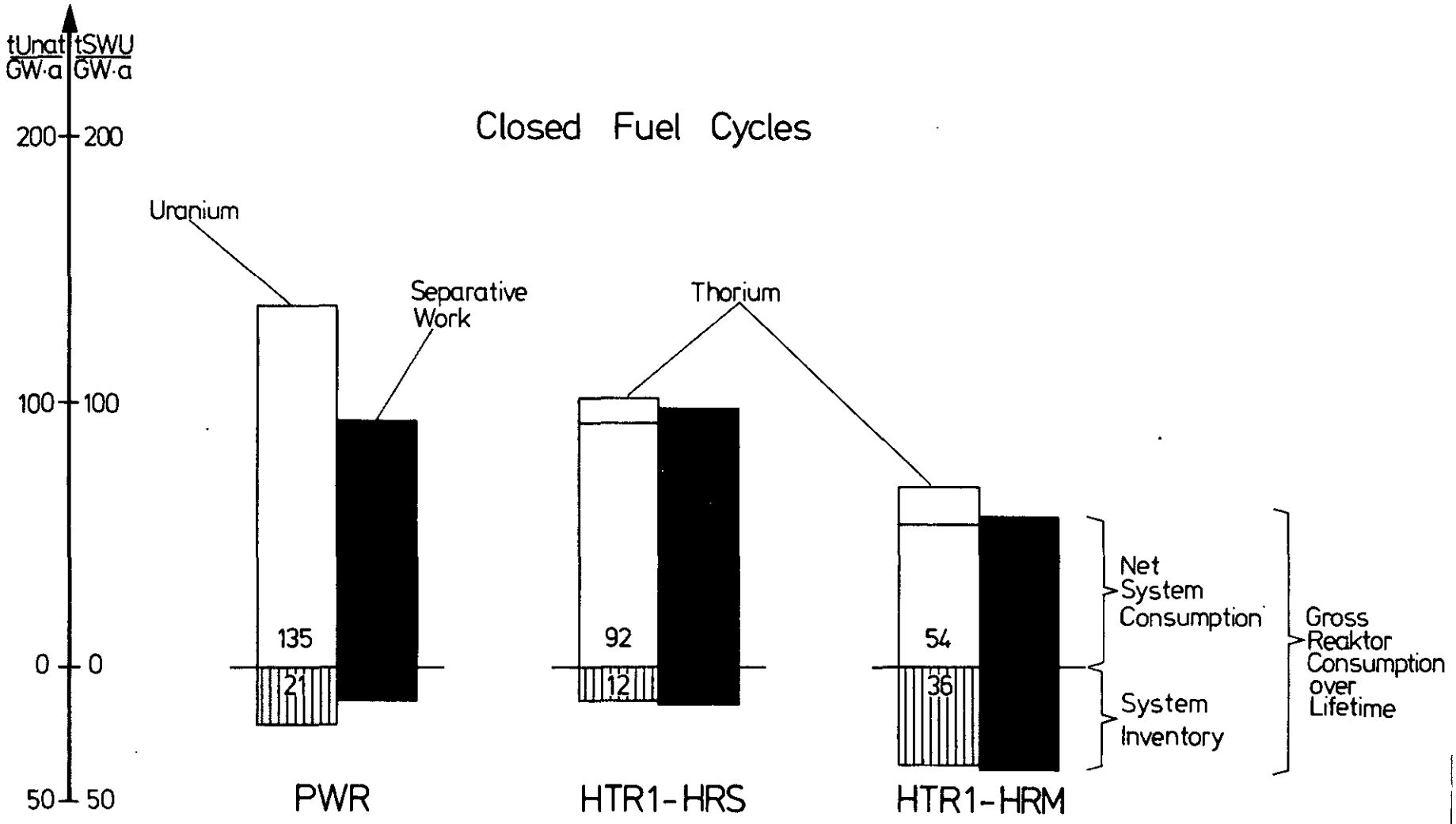


FIG. 1 Requirement for Natural Uranium and Separative Work as Average over 20 Full Power Years

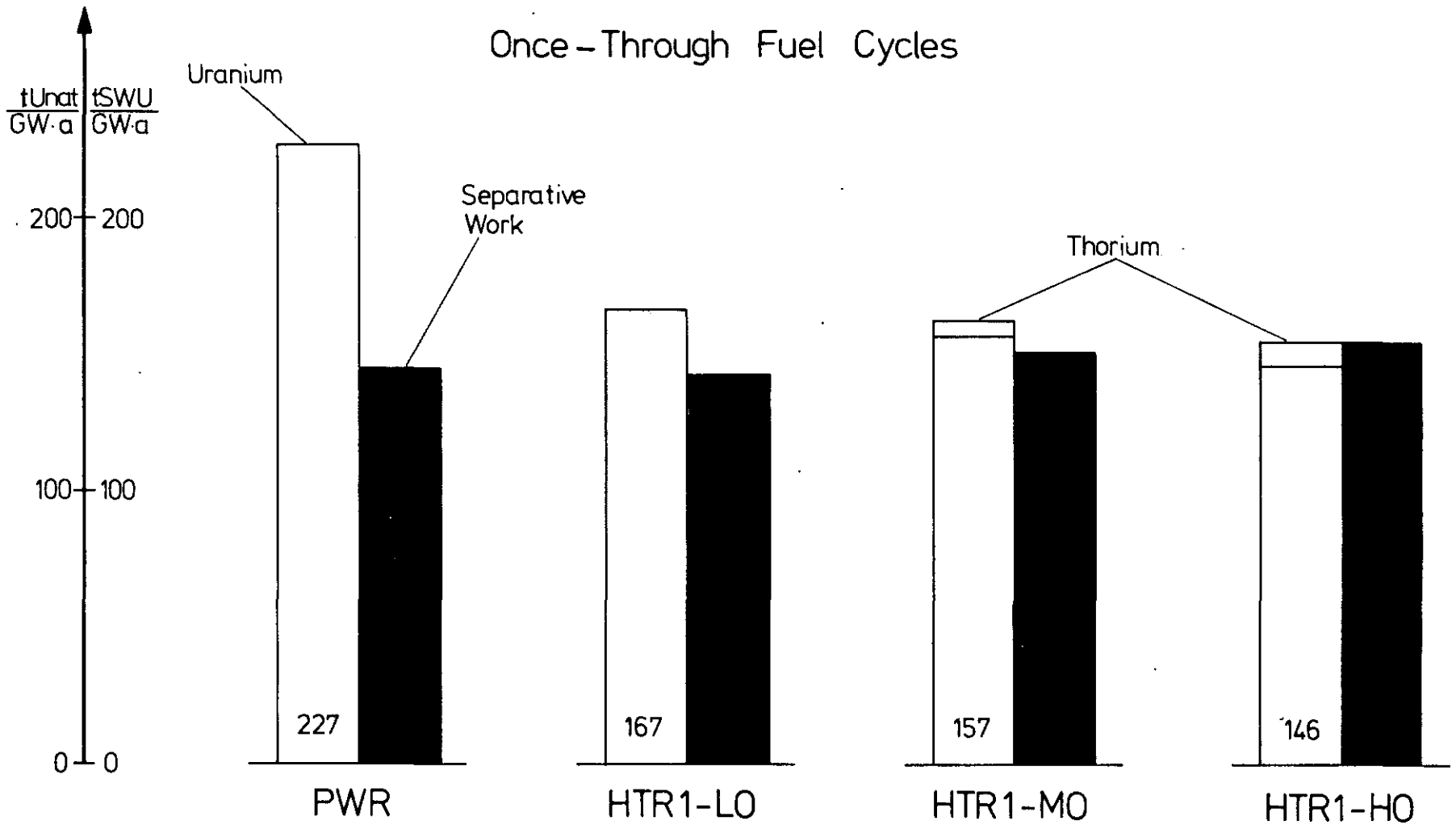


FIG 2: Requirement for Natural Uranium and Separative Work as Average over 20 Full Power Years

The HTR fuel cycle activities in the FRG

by P. Engelmann, KFA Jülich
D.F. Leushacke, KFA Jülich
G. Kaiser, KFA Jülich

The complete HTR fuel cycle consists of the following steps (fig.1)

- fabrication of fresh fuel elements
- use of the fuel in the reactor
- transport of spent fuel elements
- interim storage of spent fuel elements
- reprocessing and waste treatment
- final waste storage
- fabrication of coated particles and fuel elements from recycled fuel (refabrication)

Of these steps all but reprocessing, waste disposal and refabrication were fully developed during the past 2 decades. About 1 million spherical fuel elements for AVR and THTR containing about 25 billion coated particles were fabricated by HOBEG. Various kinds of fuel elements, containing especially mixed oxide fuel, mixed carbide fuel, low enriched UO_2 fuel, separate feed UC_2 and breed ThO_2 particles were tested under normal and transient conditions in test reactors (e.g. R2-Studsvik) as well as in statistical numbers ($\geq$3000 fuel elements) in the AVR reactor. Coatings of pyrolytic carbon (BISO) and carbon plus SiC (TRISO) were used to provide good fission product retention. In the AVR, a large number of fuel elements already has reached a burn-up of 190 000 MWd/t HM without failure. The quality of the BISO fuel is completely sufficient for HTR power plants with steam cycle for HHT and process heat reactors, however, TRISO fuel will probably be necessary for fission product retention to enable the maintenance of reformer tubes in a large process heat plant and of the gas turbine in a direct cycle HTR. Work on advanced particle concepts and on fuel elements with higher heavy metal loading is underway. We are confident, that with the AVR and THTR experience and the current r+d program, satisfactory fuel elements can be specified and fabricated also for direct cycle HTR plants and process heat HTRs.

The following 3 slides (fig. 2-4) show the HOBEG facilities for $(U/Th)O_2$ kernel fabrication, the coating furnace and the THTR fuel element production line.

After burn-up in the reactor, the fuel is stored in steel canisters - each containing 1000 fuel elements - at the site. Until reprocessing will be established, the containers after 2 years can be taken to an intermediate long term storage facility. The technical conception and

the preliminary plan of a surface storage bunker for spent fuel elements from the THTR has been established (fig.5). The capacity of the storage facility is dimensioned to collect the waste of the reactor over 10 operation years. Altogether, approx. 1,8 Million elements (i.e. 20 - 30 t HM) corresponding to a burn-up of 1,850,000 MWd can be stored. The afterheat of the fuel elements will be dissipated into the ambient atmosphere by natural convection with air. A preliminary safety report has been submitted together with the first conceptional design. The decision as to the choice of site combined with the initiation of the accompanying licencing procedure can be expected in the near future. If HTR plants will be build according to our expectations, until the year 2000 there will a total of 100 - 200 tons HM of spent fuel. Storage of this amount will cause no technical or financial problems.

If for any reason the HTR should not be successful in the market, the spent fuel elements can - from a technical point of view - be brought to final storage without any further treatment: the coatings and the graphite matrix prevent any fission product leakage and protect the kernels efficiently against chemical attack. To underline this capability the experimental underground storage of 100 000 spent AVR elements in the salt mine ASSE has been prepared in all its steps - including transportation from Jülich to the ASSE near Braunschweig. Fig. 6 shows the transport cask being lowered to a borehole in the salt. Each borehole will be filled with a stack of 25 canisters containing a total of 25 000 fuel elements and will then be closed by a concrete block. The storage of the first canisters was planned for april 1977 but had to be postponed because of public opposition and is now planned for later this year.

Our reference fuel cycle concept, however, goes towards reprocessing and refabrication, because of the better fuel utilization. After cold and hot laboratory scale testing of all reprocessing steps at Jülich, a reprocessing pilot plant, called JUPITER (JUelich PIlot facility for Thorium Element Reprocessing) is now under construction. Fig. 7 gives a simplified process flow diagramm of this facility, which will have a capacity of 2 kg heavy metals per day.

The process employed consists of the following steps:

- size reduction of spherical fuel elements in a hammer mill
- removal of matrix graphite and PyC-coating by fluidized-bed burning
- dissolution of the exposed fuel particles in fluoride-catalyzed nitric acid
- feed adjustment by evaporation and steam-stripping of nitric acid
- recovery and decontamination of thorium and uranium by TBP solvent extraction

The main processing steps outlined before are supplemented by other unit operations suchs as

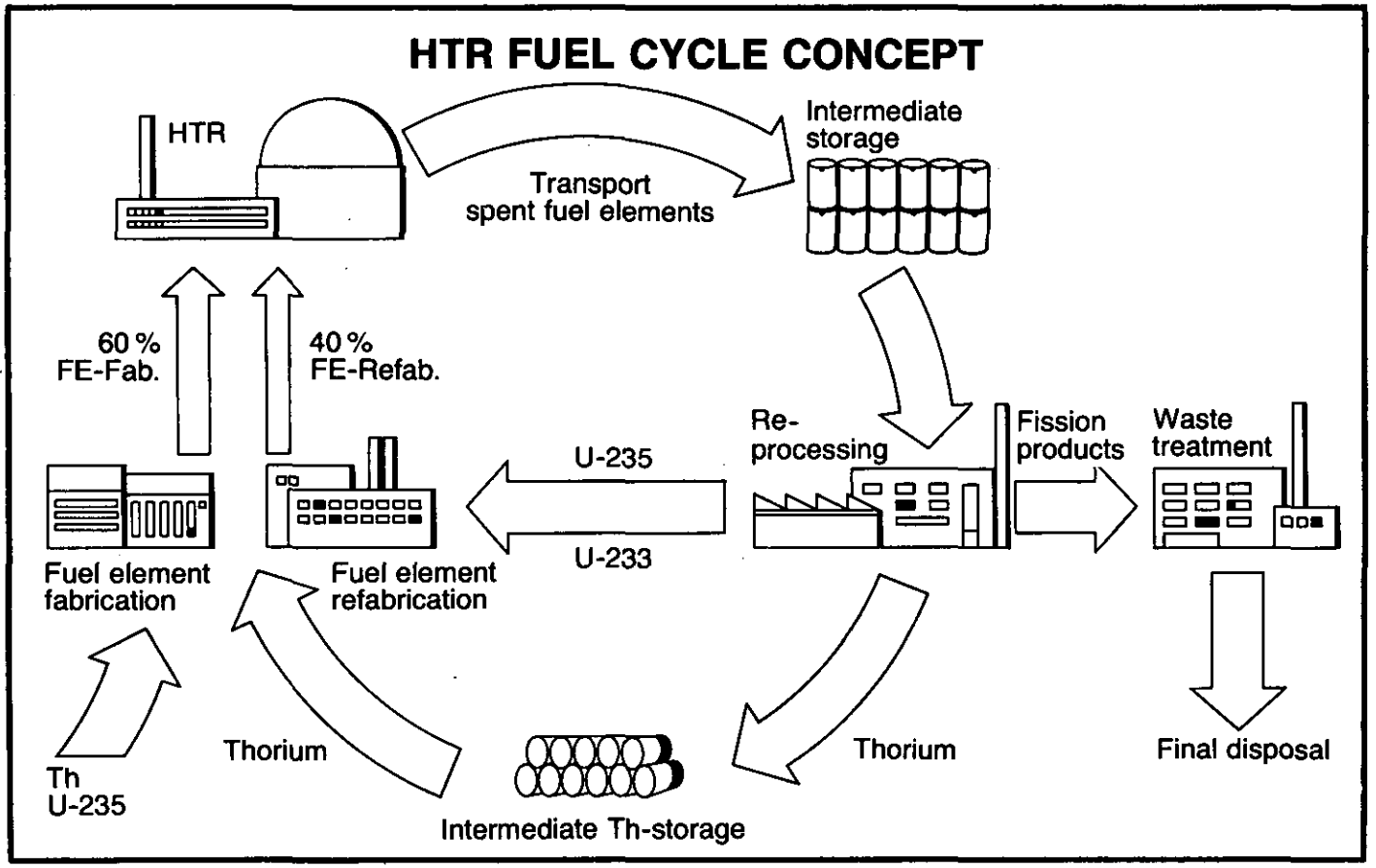
- nitric acid recovery
- solvent regeneration
- burner, dissolver and vessel off-gas cleanup ect.

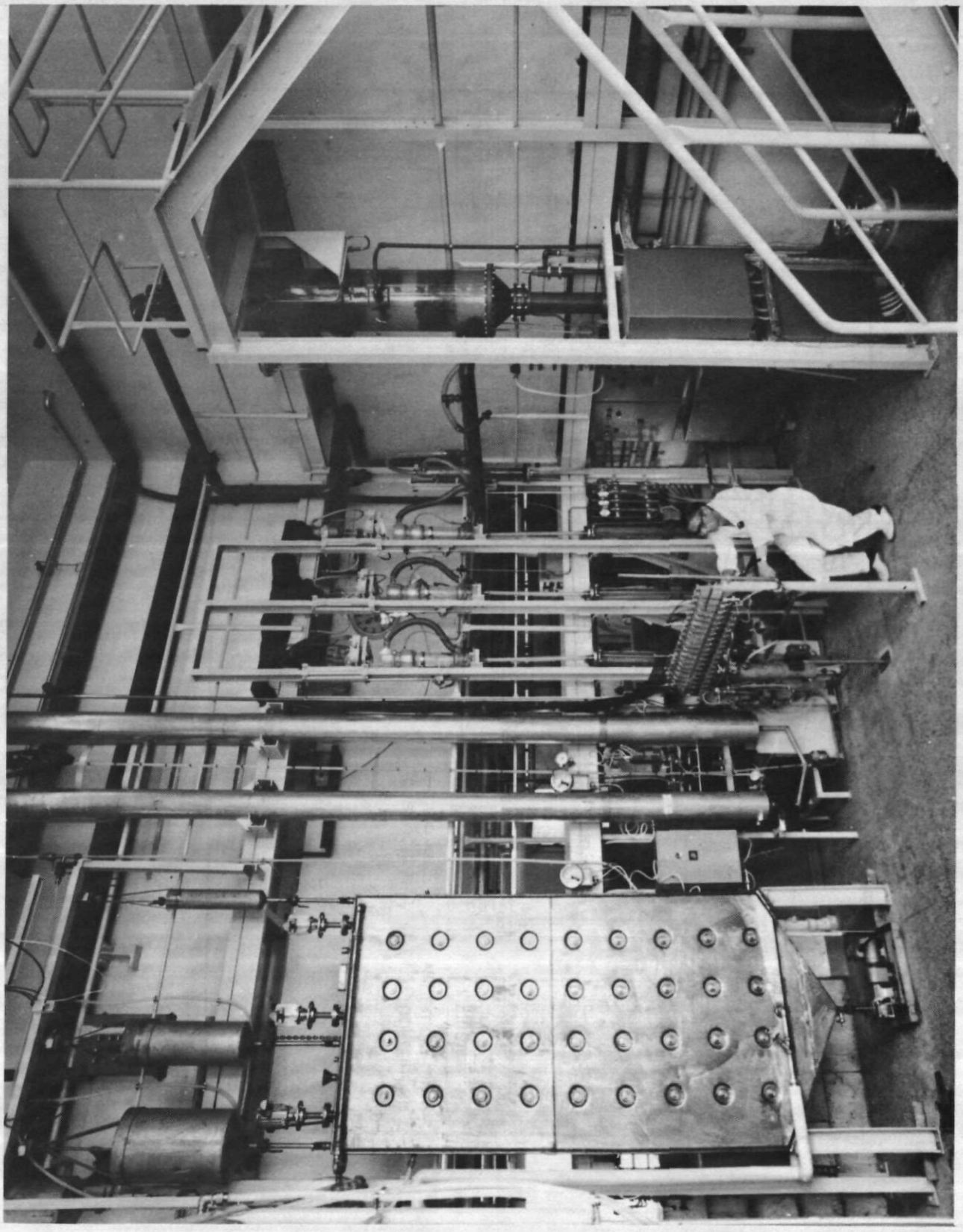
which are required for economic and ecological reasons.

The entrance cell (fig. 8) and the head-end cell with the hammer mill and fluidized-bed burner (fig.9) are complete; installation of the chemical processing cell, which is the most complex part, will be started in spring this year. Checkout and shakedown of the first two cells is scheduled for 1978, start-up operation of the full line is expected in 1980.

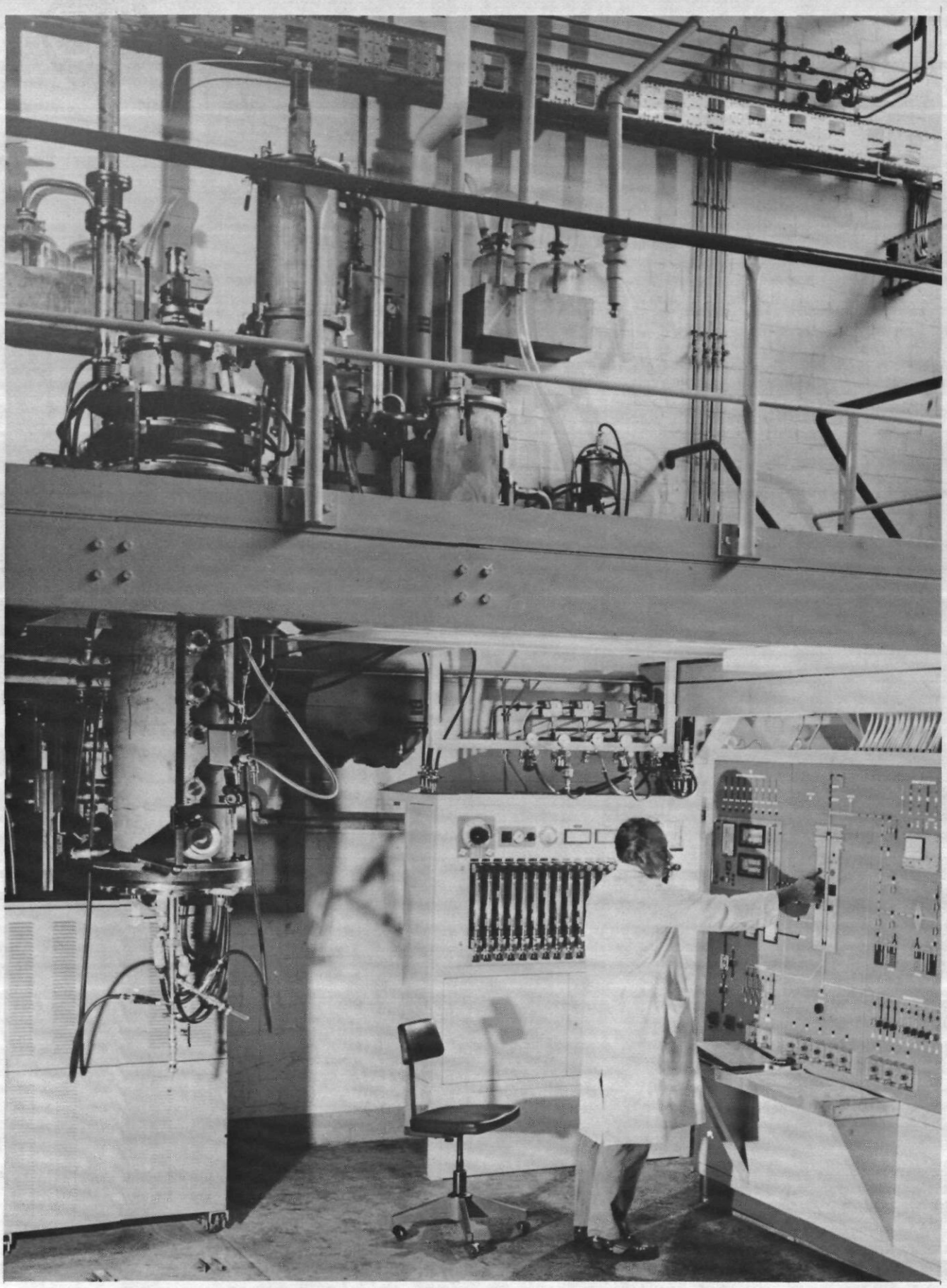
The waste treatment for JUPITER has already been developed. A small plant (FIPS I: Fisson Product Solidification) with a throughput of approx. 1 kg glass/h has been put into hot operation. The specific activity of the glass came up to 7000 Ci/kg. The experimental program was finished end of 1976 and investigations of the components (corrosion effects) are in progress. An improved facility with a higher throughput and for continuous operation, FIPS II, is under construction.

Our plans foresee, that in the nineties a larger demonstration facility for reprocessing and refabrication can be designed and built in cooperation with the US DOE. Our activities in Germany and the fuel recycle activities at ORNL, GAC and INEL were planned in such a way that both programs are complementary and can be merged to a common program. A great number of joint activities have been already started forming a sound basis for a technical program to close the HTR fuel cycle in both countries. Looking at the time scale of a possible market introduction of the HTR, commercial reprocessing and refabrication should be available by 2010. Our r+d programs in the area of the fuel cycle can easily meet this date. Nevertheless we consider the establishment of the full HTR cycle as an important task. Intermediate solutions must be available to encourage the utilities until they will order the first plants. We have therefore organized in 1977 all HTR fuel cycle activities at KFA and in German industry in the HBK-project (Projekt Hochtemperaturreaktor-Brennstoffkreislauf) under the leadership of KFA to strengthen our r+d activities and to prepare for an intense international cooperation in this area.

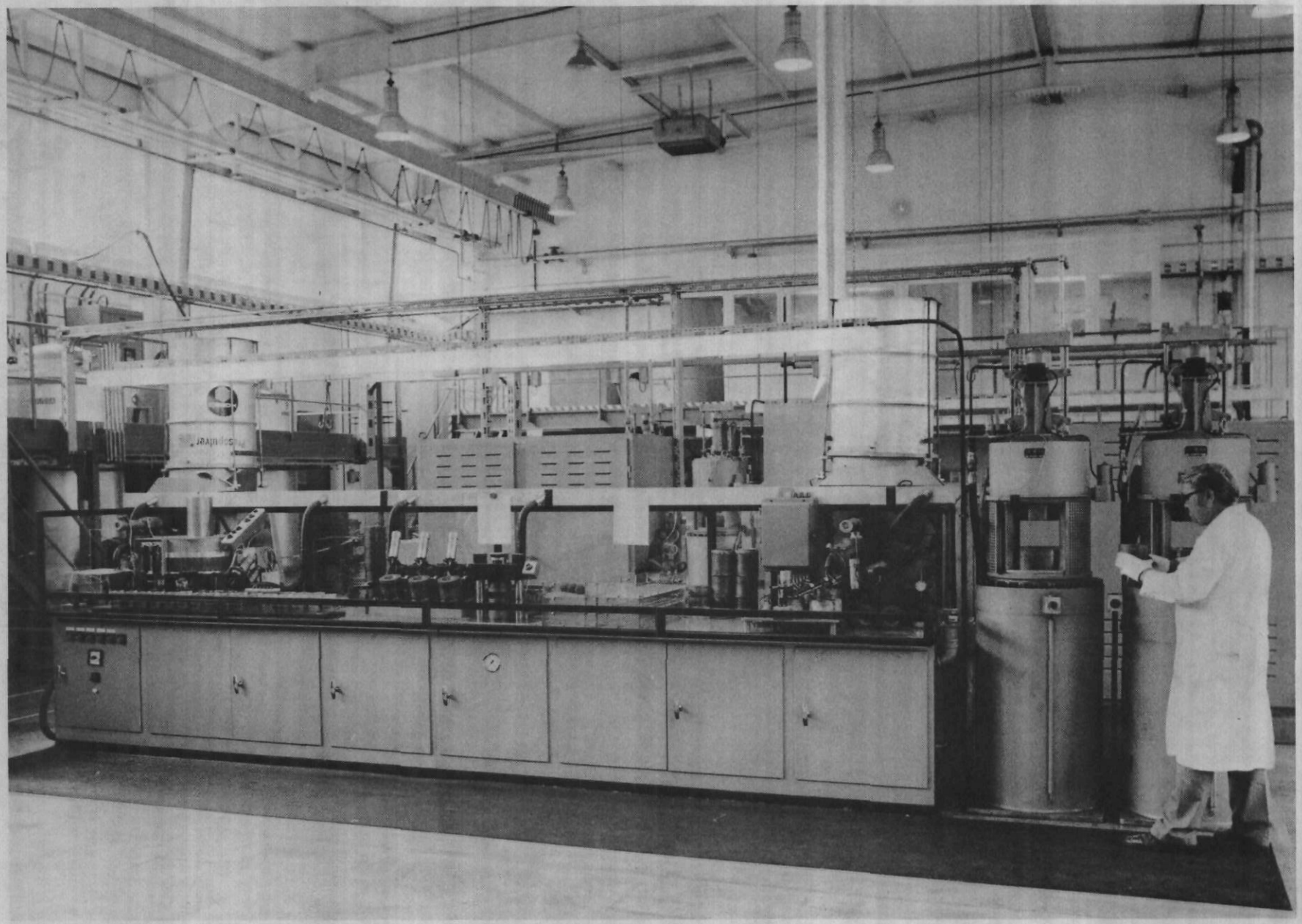

HTR FUEL CYCLE CONCEPT
HTR
Transport
spent fuel elements
Intermediate
storage
60 %
FE-Fab.
40 %
FE-Refab.
Re-
processing
Fission
products
Waste
treatment
U-235
U-233
Fuel element
fabrication
Fuel element
refabrication
Th
U-235
Thorium
Thorium
Final disposal
Intermediate Th-storage



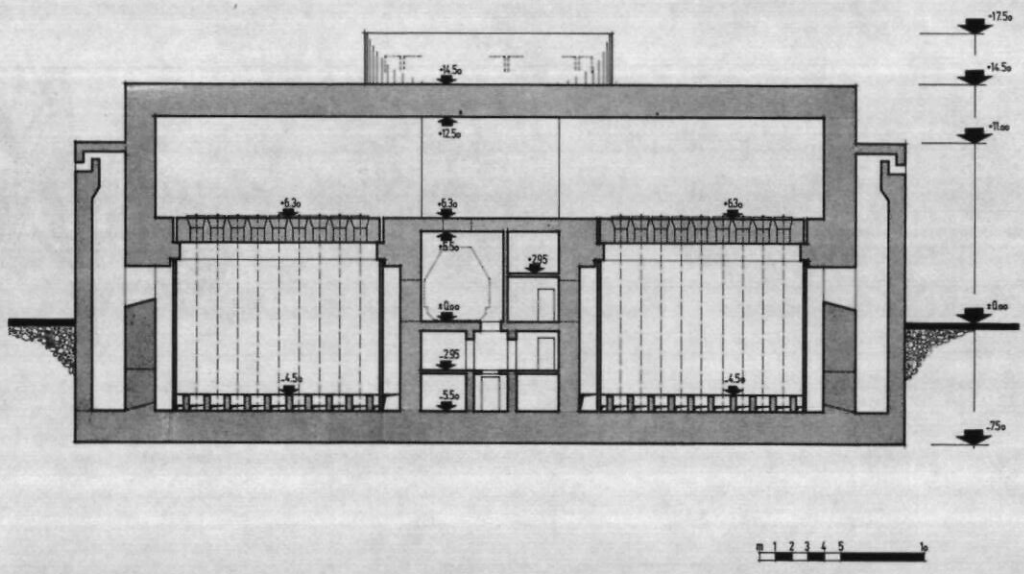
$(U/TH)O_2$-KERNEL FABRICATION

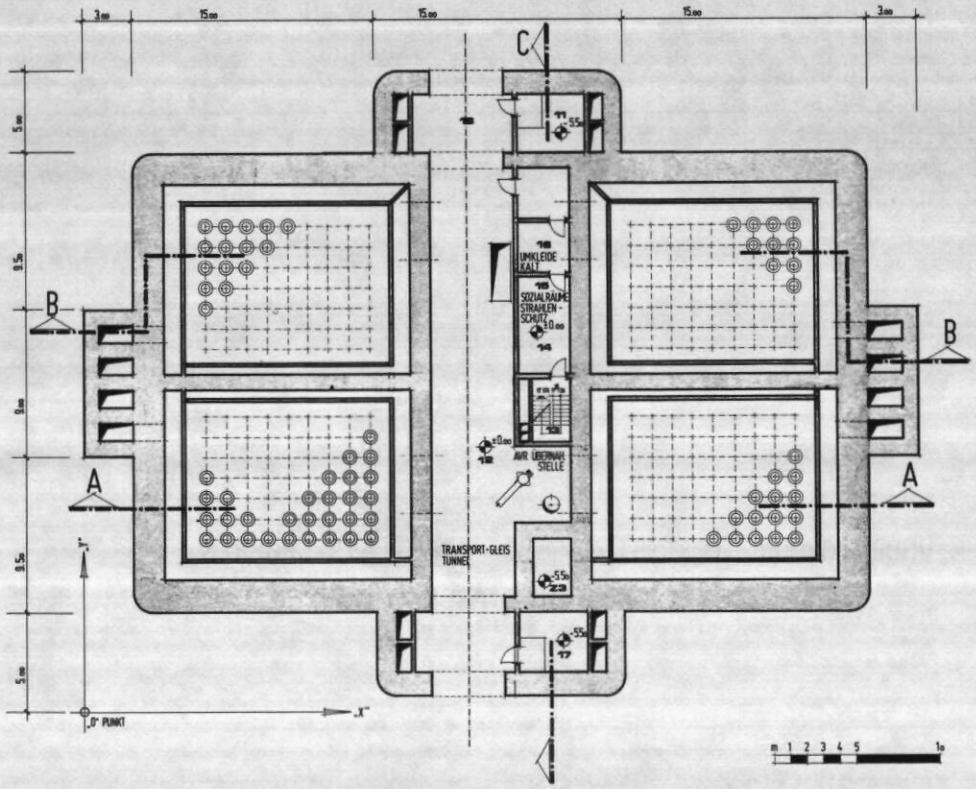


PARTICLE COATING FURNACE



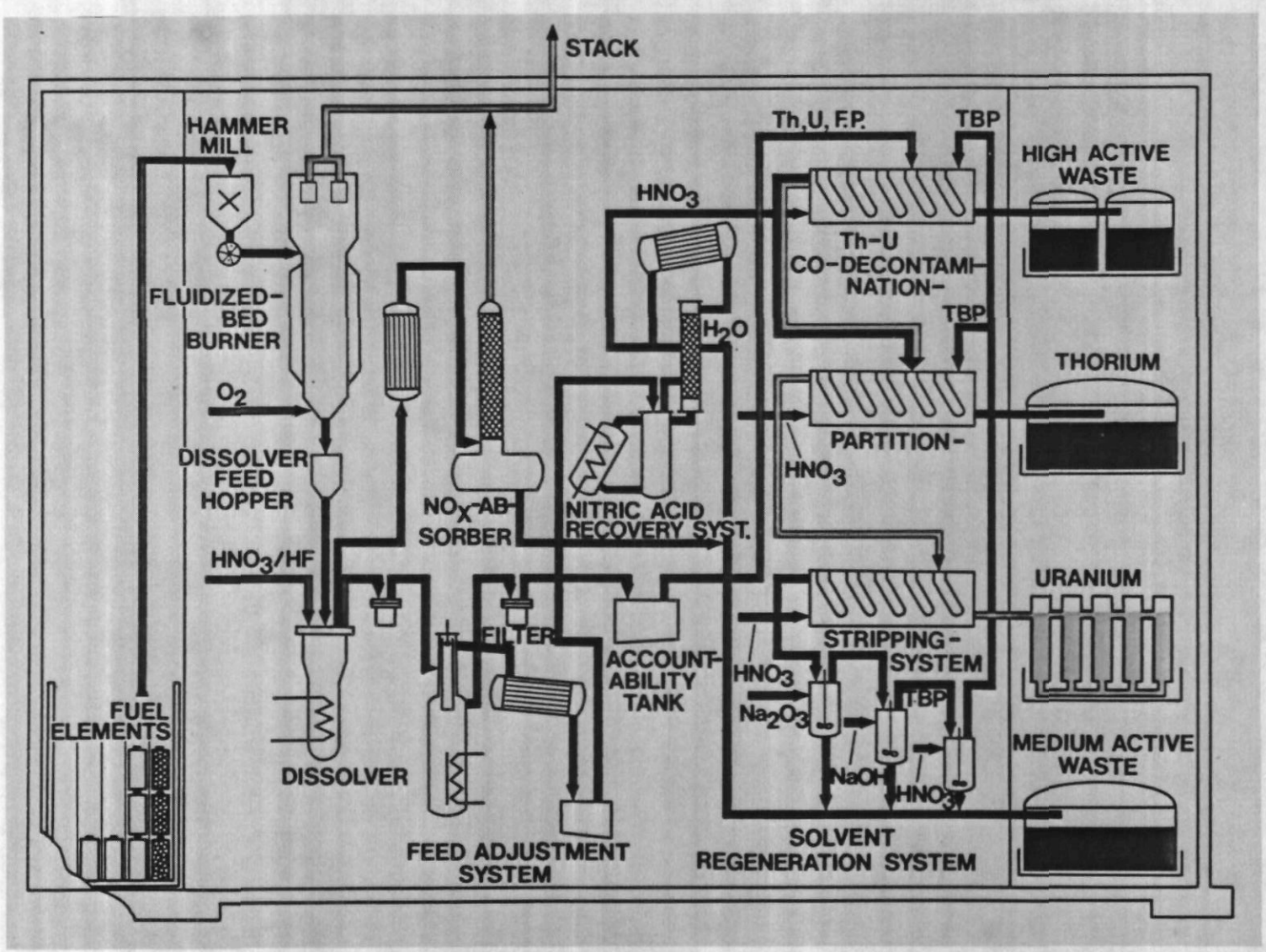
THTR FUEL ELEMENT PRODUCTION LINE



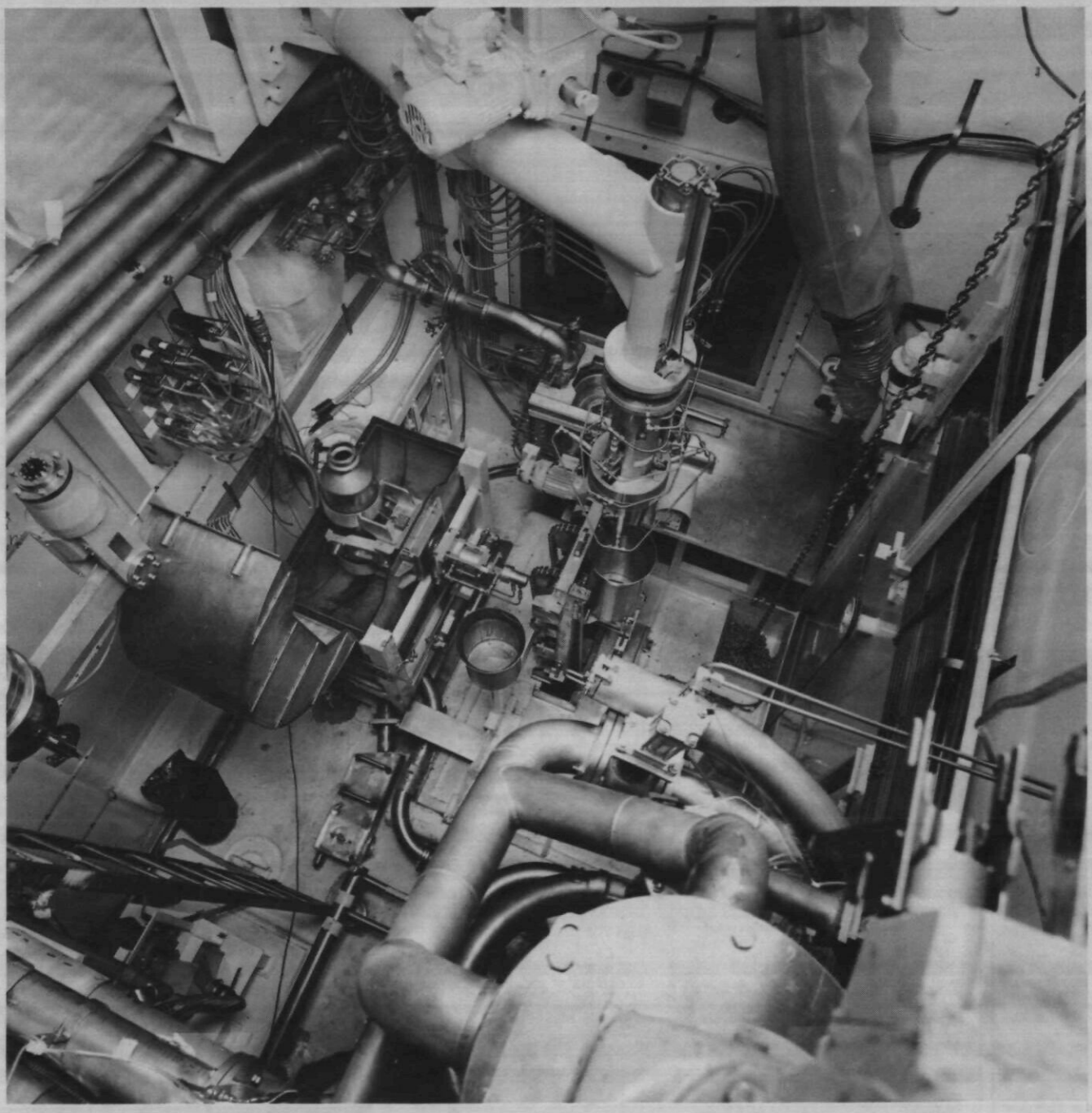


CONCEPTIONAL DESIGN OF THE SURFACE STORAGE BUNKER
FOR SPENT THTR FUEL ELEMENTS

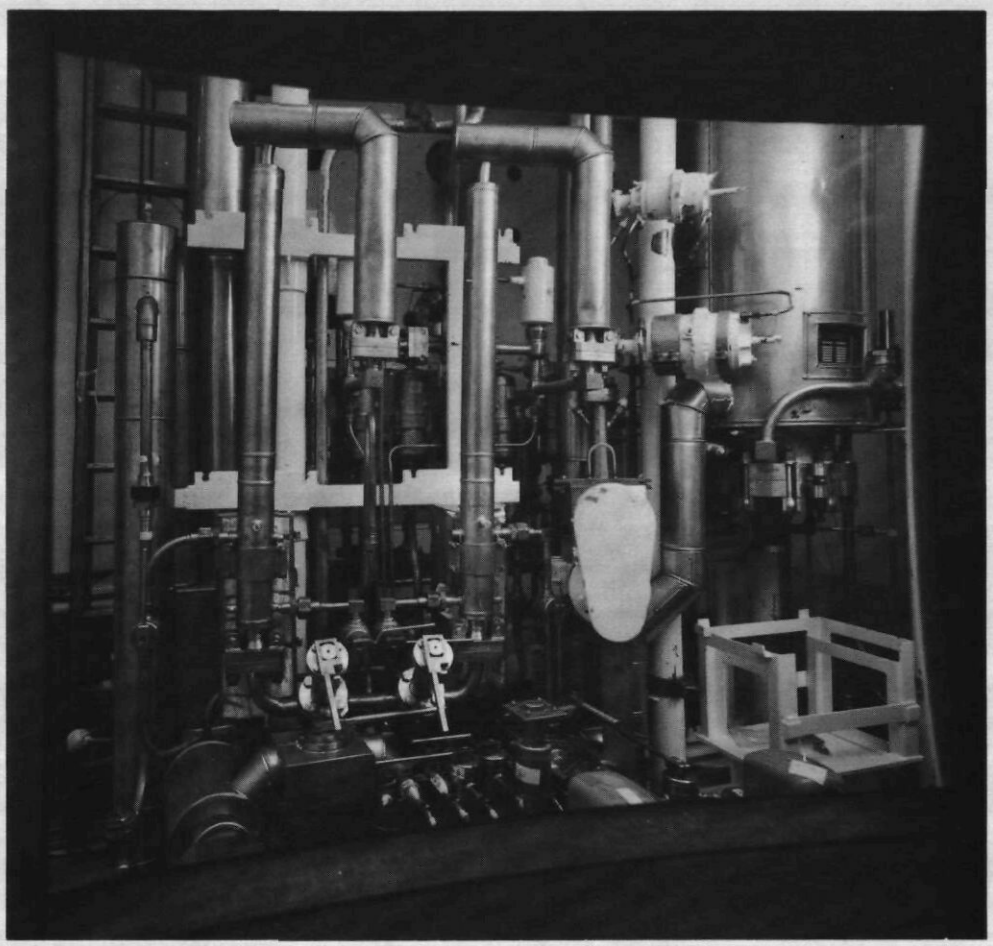
UNDER-GROUND TEST STORAGE OF SPENT AVR SPHERES
IN THE SALT MINE ASSE



SIMPLIFIED PROCESS FLOW DIAGRAM OF THE JUELICH PILOT PLANT FOR THORIUM ELEMENT REPROCESSING



PICKING PLANT FOR SPENT AVR SPHERES



JUPITER HEAD END CELL
Lower Part of Fluidized Bed Burner

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Jülich, den 9. 1. 1978

Some Remarks on the Safety of HTRs

The safety of nuclear power reactors is a fairly complex matter which is not easily definable. It comprises such diverse aspects as a low emission of radioactivity and a low frequency of troublesome incidents during normal operation, an acceptable average risk, and acceptable consequences even in the case of a very severe accident, however improbable this may be.

In a paper like this it is not possible to cover all aspects of HTR safety. In Sweden, there are excellent and comprehensive accounts on this subject both in Swedish and English by Ekholm, Runfors and Vieider. I shall try to consider some of the typical HTR safety properties from particular points of view.

As no commercial HTR-plants of the size which is usual today have been realized, we shall have to discuss mainly the generic safety properties, i.e. those which are common to all HTRs. But I think we cannot avoid at certain points to consider specific planned plants, especially the HTGR-1160 of the GAC, in order to obtain quantitative statements.

1. Normal Operation

Experience with the normal operation of nuclear power plants in the last 10 years has shown that even minor incidents are the cause of great anxiety and public discussion. For the practical operation and the continued availability of power reactors it is therefore very important to avoid such incidents.

In normal operation the HTR shows a high stability and very good self-regulating properties. The high heat capacity of the graphite moderator and the comparatively low power density retard all temperature transients whatever their cause may be (Fig. 1).

Reactor Type	Reference Plant	Power Density [MW/m³]
Pressurised Water Reactor	Biblis B	92
Boiling Water Reactor	Krümmel	51
Fast Breeder Reactor	SNR-300	375
High Temperature Reactor	THTR 300	6
	HTR 1160	8,5

KFA ISF — Average Power Density of Different Reactor Types — 22.12.77

Fig. 1

Short time transgressions of the normal operational values of fuel and coolant temperatures do not create safety hazards. The permissible fuel temperatures are determined by the long time behaviour of the fuel, i.e. they are limited to such values that even after several years of fuel irradiation the fission products are still safely retained. Even considerable increases in temperature do not result in any abrupt or irreversible changes of the physical properties of core components such as melting or evaporation (Fig. 2).

Material	Operating Temperature [°C]	Failure Temperature [°C]
Graphite	770	3600
UO_2	1100	2800
Coated Particle	1100	1800
Shut Down Rod	600	1350
B_4C	600	2400
Insulation (upper)	340	815
Liner	50	950

KFA ISF — Characteristic Operating Temperature and Failure Temperature of Specific Reactor Components — 20.12.77

Fig. 2

An impressive demonstration of the large safety margin was the increase of the coolant outlet temperature of the AVR from the original design value of 850° C to 950° C which proved to be possible without any major changes to the system.

The value of the radioactive emissions, which are the bases of the licencing procedures for the HTRs which are presently being developed or are under construction, are lower than the emissions of existing power plants of a comparable size (Fig. 3).

Reactor		THTR-300		HTR-1160	
		Design Values	Expected Values	Design Values	Expected Values
Atmospheric Discharge	Noble gases	25000 Ci/a (continuously) 6000 „ (discontinuously)	2000 Ci/a	1000 Ci/a (continuously) 5000 „ (discontinuously)	160 Ci/a
	Tritium	10 „		10 „	
	Iodine (+Cs + Sr)	10^{-4} „	$\sim 10^{-6}$ „	$4 \cdot 10^{-3}$ „ (without Filter)	$\sim 3 \cdot 10^{-6}$ „ (with Filter)
Liquid Discharge	Tritium	1000 Ci/a	400 Ci/a	2000 Ci/a	800 Ci/a
	solids (mainly Fe55 and Fe59)	0,5 „	0,2 „	0,5 „	0,2 „

KFA ISF — Radioactive Emission from High Temperature Reactors — 2112 77

Fig. 3

The values originate from calculations based on conservative models and data from fission product release experiments. Considerably lower values are obtained by extrapolating the measured emissions from experimental HTRs. In principle the emissions may be further reduced by improving the gas purification plant and the fission product retention in the coated particles. This also holds for tritium which is considered by some to be problematic.

Safety problems could arise from maintenance work on the primary circuit, especially the steam generators. A part of the gasborne radioactivity circulating in the primary circuit plates out on the walls of the circuit or on components and partly diffuses into the material particularly in those parts of the circuit operating at high temperature. Judging from the experience gained with small HTRs operated so far, it is to be expected

that radiological doses resulting from maintenance and repair work are well below the doses which are regarded as acceptable today. Moreover, a further reduction of the contamination of the primary circuit by improving the retention of fission products in the fuel seems both possible and desirable.

2. Design Basis Accidents and Component Safety

All power reactors are designed in such a manner, that Design Basis Accidents do not cause any serious damage to the public. Design Basis Accidents therefore are not very well suited for a judgement of the safety of a specific type of plant.

For HTRs three different types of DBAs are discussed:

- The Depressurization Accident, sometimes not quite correctly referred to as Loss of Coolant Accident,
- The Air Ingress Accident,
- The Water Ingress Accident.

The Design Basis Depressurization Accident (DBDA)

The fast depressurization of the primary circuit is the counterpart of the classical loss of coolant accident of an LWR. For a modern design of an HTR such as for example the GA-1160, which has a prestressed concrete reactor vessel and podboilers, a double ended rupture of the primary circuit is physically impossible. What has to be envisaged is the failure of the closure of one of the external openings of the primary circuit at the surface of the PCRV. Design features such as flow restrictors are introduced to ensure that in this case the maximum free cross section does not exceed a few hundred cm^2. This is to prevent damage to any core components by the pressure gradients which are created in the primary circuit and in the core. But even much larger openings do not necessarily result in the failure of core components as in this case the speed of depressurization is mainly determined by the flow resistance of the circuit itself (Fig. 4).

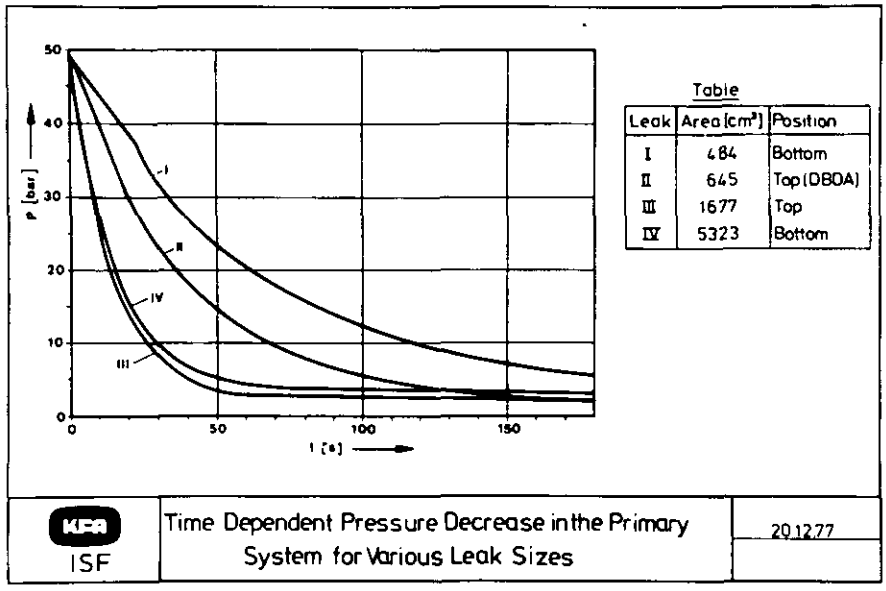
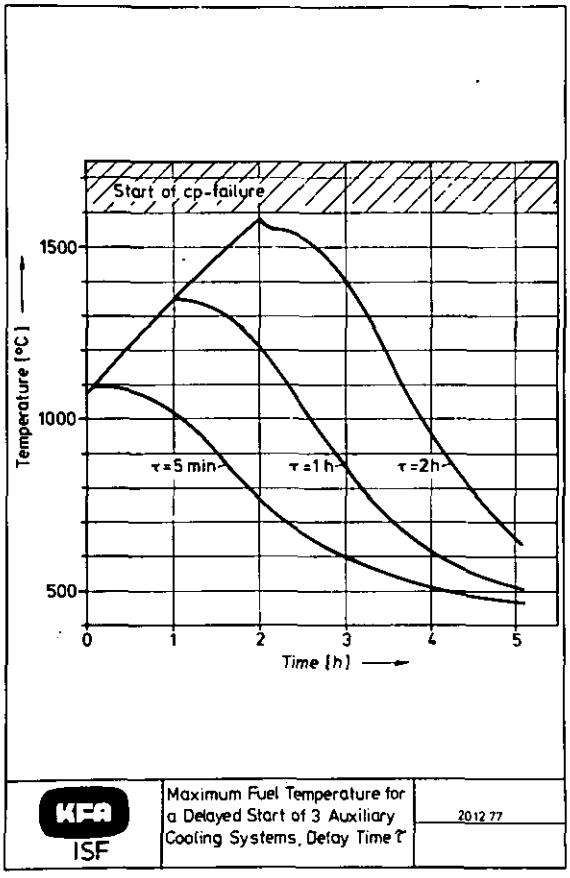


Fig. 4

Physically the depressurization is a simple process which can be well calculated using simple physical models and computer codes, as there are no phase changes and no two-phase flow phenomena. To a first approximation, the exact position of the leak does not influence the consequences. A ruptured liner would not cause greater problems than a failed closure, although the effects of hot gas streams must, of course, be considered in the layout of the plant.

The gas which is left in the primary circuit after a depressurization is sufficient to remove the after heat from the core using the normal blowers and steam generators. The efficiency of the primary cooling circuit is decreased to a few percent as a result of the pressure loss, but so is the heat production. This, together with the high heat capacity of the core which slows down the temperature rise of the core to a few degrees Centigrade per minute, are the reasons why the HTR does not need an emergency core cooling system nor a separate after heat removal system, and in fact, the Fort St.-Vrain Reactor and the THTR-300 possess neither.

The further course of a depressurization accident strongly depends on the efficiency and reliability of the after heat removal. With an HTR the efficiency of this system can easily be tested under accident conditions, and this actually is a part of the commissioning schedule and periodic testing. The demands on the reliability of the system are not as high as with LWRs. As the core heats up very slowly it is quite sufficient if the cooling is restored after an hour or two (Fig. 5).



As the failure rate of the after heat removal system is dominated by the failure rate of the power supply, this leaves ample time for counter measures.

The radiological consequences of a DBDA are comparatively modest. Only a part of the fission product contents of the primary circuit is liberated into the containment, the large core inventory remaining unaffected.

Fig. 5

The Design Basis Steam and Air Ingress Accidents

The Design Basis Steam Ingress Accident analysis is concerned with the effect of a sudden off-set rupture of one tube in that part of a steam generator which leads to the maximum ingress rate of water and steam. The analysis must take into account:

- the resulting pressure transient
- the immediate effects on fission products external to the coated particles
- the possible formation of flammable gas mixtures
- the possible positive increase in the nuclear reactivity.

These are included in the kinetic analysis carried out for the G.A.C. 1160 design, for example.

The graphite-steam reaction is stopped by the cooling of the shut-down core, heat being removed by the main or the auxiliary loops and by the endothermic reaction itself. The supply of water is cut off by water monitors, backed by pressure and flow sensors, acting to close the faulty secondary

system. Even in the case of the longest delay in this action, the release of gas to the containment through the pressure relief valves contains not more than about twice the gaseous activity of the DBDA release, the concentrations of H_2 and CO are far below the flammability limit and the positive reactivity increase is trivial.

Air ingress is potentially more dangerous because the reaction is much faster and is also exothermic. However air can enter the primary circuit only by convective gas exchange following a depressurization, i.e. with the reactor shut down and cooling in progress.

The conclusions of the analyses of this range of accidents are that all the oxygen which enters the primary circuit in the first hours will be converted into carbon monoxide but that at no time will the reaction heat be more than a few per cent of the decay heat and the total will never exceed the cooling capability. Sustained combustion is therefore impossible.

The concentrations of flammable gas entering the containment by exchange is below the flammability limit of the mixed gas inventory by more than a factor of 1000. A deflagration is conceivable only if one assumes the formation of separate layers of flammable gases and containment atmosphere. Interest in the investigation of these phenomena has arisen because the only realistic cause of a containment failure would be a gas explosion.

The Prestressed Concrete Reactor Vessel

I should like to add a few remarks on the Prestressed Concrete Reactor Vessel (PCRV) which to a large extent determines the inherent safety properties of the HTR. The primary coolant gas is contained by a comparatively thin walled steel vessel, the so-called liner, which transmits the forces exerted by the gas pressure to the concrete. As the liner is not exposed to internal strains and is operated at temperatures of about 50° C, the probability of a liner rupture is very low. The internal stresses in the concrete are taken up by a highly redundant system of several hundred vertical and circumferential steel tendons. The design also takes into account so-called pressurized cracks, so that in normal operation a considerable safety margin exists. The tendons are not exposed to radiation or to high temperatures and may be easily inspected. It is therefore generally accepted that a catastrophic failure of the PCRV under normal operating conditions can be excluded.

The critical points of a PCRV are the plugs closing the large openings of the PCRV. If the fastening of a plug fails, the plug may be accelerated to such a velocity that the containment may be damaged. For future PCRV's the plug will be integral with the pre-stressing system of the main vessel.

3. AIPA-Study

It is only a few years since the investigation of reactor accidents has been extended beyond the range of the Maximum Credible Accident. In looking at the wide field of the so-called hypothetical accidents, one is bewildered at first by the large number of conceivable accidents and by the seemingly impossible task to investigate them all. The most rational method of dealing with this problem is the probabilistic risk analysis, which determines for all possible types of accidents their respective occurrence probabilities and their consequences. The product, which has the dimension of an average expected damage per year, gives the risk associated with each type of accident. It turns out that only a few types of accidents contribute significantly to the total risk so that only these accident sequences need to be investigated in detail.

The usefulness of this method has been demonstrated with LWRs in the well known Rasmussen Study. A similar study for the HTR-1160, the AIPA Study, was published in 1976. The main results of this study are: The risk caused by an HTR is extremely low and mainly due to three different types of accidents, namely

- a Loss of Offsite Power,
- a Leak in the Reheater,
- an Earthquake far exceeding the Safe Shut Down Earthquake.

The mechanism by which a loss of offsite power may develop into an accident is not only characteristic for an HTR, but also plays an important role with other reactors, too. If the offsite power is lost, the electrical output of the plant must be rapidly reduced to the power consumption of the plant itself. This may cause the turbine to trip, so that the reactor must be shut down.

The reactor then depends on the diesel generators for the supply of the electric power needed for the core auxiliary cooling system. If the diesel generators do not start, the consequence may be an unrestricted heat up of the whole core eventually resulting in a massive liberation of fission products into the containment. A failure of the containment as a consequence of this accident is not conceivable.

A core heat up accident may be initiated by other events, too, e.g. by a severe earthquake. It may be considered as the equivalent of the core melt accidents typical for reactors with a liquid coolant.

The reheater leak accident is typical for a specific HTR design. If the design pressure of the steam in the reheater is lower than the coolant pressure in the primary circuit, coolant may penetrate into the secondary cooling circuit and be released to the atmosphere. As only a small part of the gasborne activity of the primary coolant may escape, the radiological consequences remain moderate.

The AIPA Study presents its results in the form of a probability versus consequence diagram, where the consequence is defined as the individual whole-body gamma dose received in thirty days at the outer boundary of the zone of low population density. For all accidents having a probability of more than 10^{-9}/a - below this limit the method becomes questionable - the consequences are so low that immediate somatic effects are not to be expected. There are no early casualties. The risk - expected dose per year - is several orders of magnitude smaller than the permissible doses per year for normal operation (Fig. 6, 7).

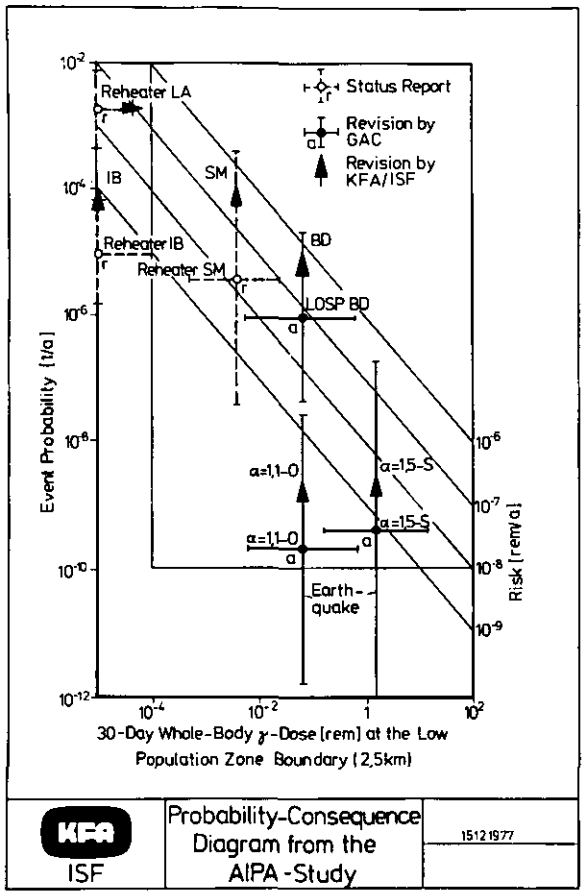


Fig. 6

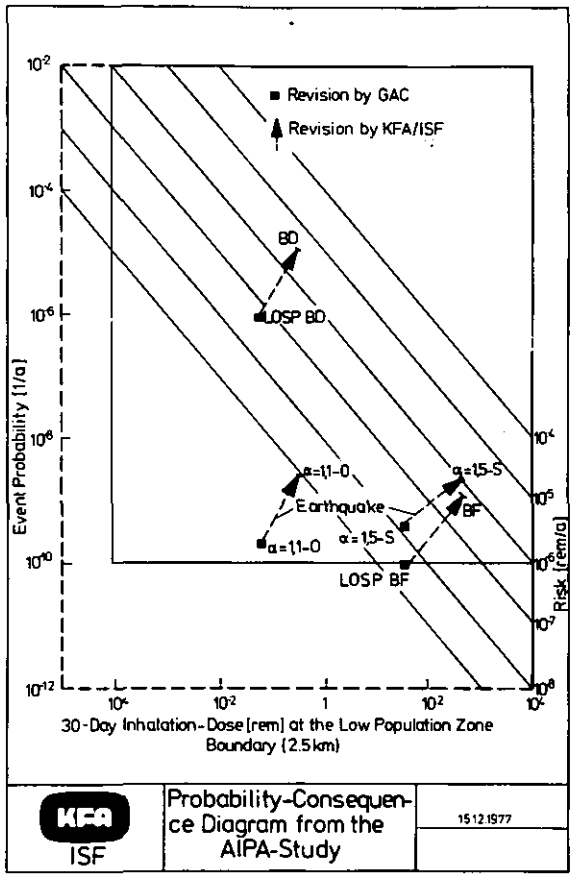


Fig. 7

The excellent results of the study have of course incited a lot of criticism. It has been argued that occurrence probabilities were too low, plate out factors were too high, and that some types of accidents had been omitted altogether. A part of this criticism has been accepted by GAC and taken into account in revised versions of some diagrams and tables. We have also thoroughly reviewed the whole study and proposed a number of modifications which would slightly increase the overall risk. However we could not detect systematic faults or errors which would justify serious doubts about the validity of the study.

Looking for the reasons of the very low total risk, one finds two major causes:

- The slow process of fisson product release gives competing processes such as decay, absorption, plate out, and chemical reaction a chance to decrease the concentration of fission products, especially of the iodine isotopes.

- The low amount of energy released during a depressurization accident and the integration of the primary circuit into the PCRV make a containment failure as a result of a depressurization accident a very unlikely event.

4. Hypothetical Accidents

A probabilistic risk analysis investigates hypothetical accidents with an extremely low occurrence probability only so far as to make sure that they do not contribute significantly to the total risk. However, there is ample

justification for the detailed analysis of at least some characteristic hypothetical accidents. Such an investigation not only improves our understanding of the basic safety properties of a reactor type but it also provides reasonable answers to some of the questions which are permanently raised by the public.

It is not easy to select one of the many conceivable types of accidents for a closer investigation without exerting some personal judgement. We decided to investigate the accident which may develop from a fast depressurization if all engineered safeguards of the reactor fail totally. Although such an accident is extremely unlikely, its consequences may be considered as an upper boundary to the consequences of accident sequences developed in a mechanistic manner (Fig. 8).

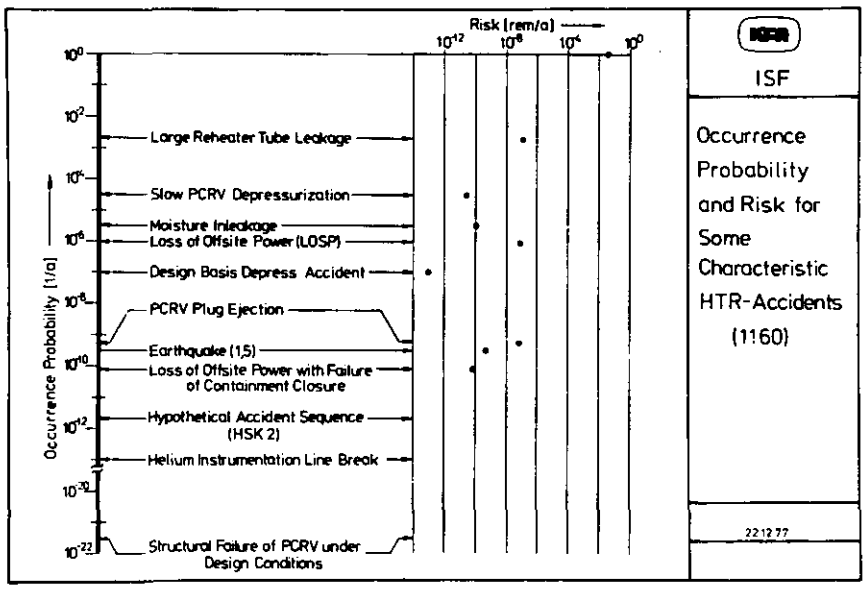


Fig. 8

The accident is initiated by a sudden leak of about 0.5 m^2, i.e. 8 times the design leak, in the closure of a steam generator pod at the bottom of the PCRV (Fig. 9). This results in a fast depressurization of the primary cooling circuit, sweeping a part of the activity contained in the primary circuit into the containment. It is assumed that the shut down system fails, that the main cooling circuits shut down immediately and that the auxiliary cooling system fails to start. This results in an unrestricted core heat up

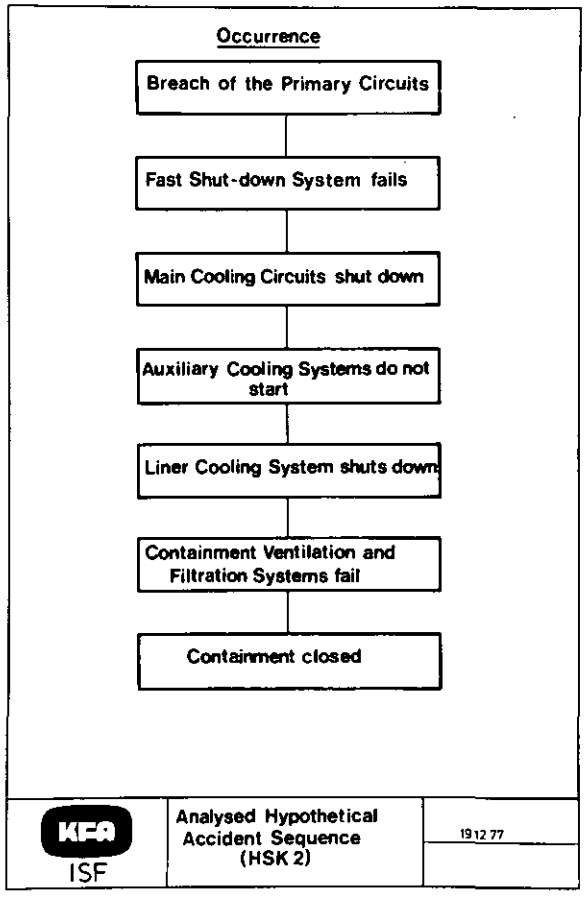


Fig. 9

which shuts the reactor down after a few seconds. As the temperature rises, core components begin to fail as a result of reaching their temperature limits (Fig. 10).

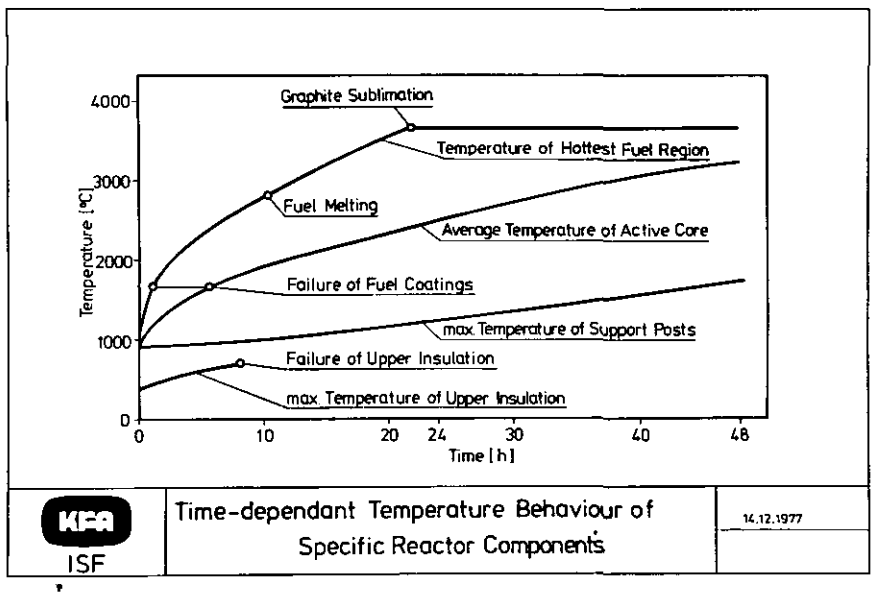
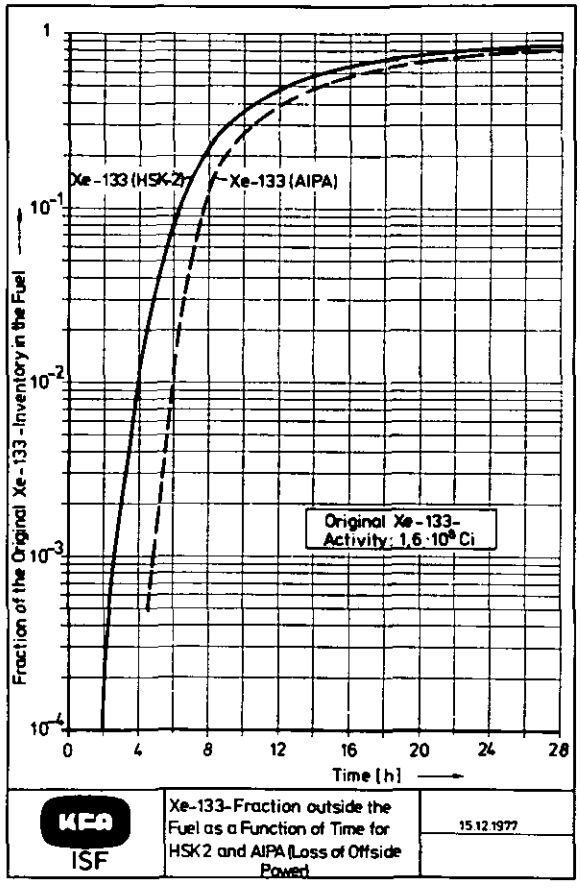


Fig. 10



The coatings of the coated particles fail between 1600° and 2000° C, releasing the fission products into the primary circuit, where a considerable part plates out in the cooler parts of the PCRV (Fig. 11, 12).

Fig. 11

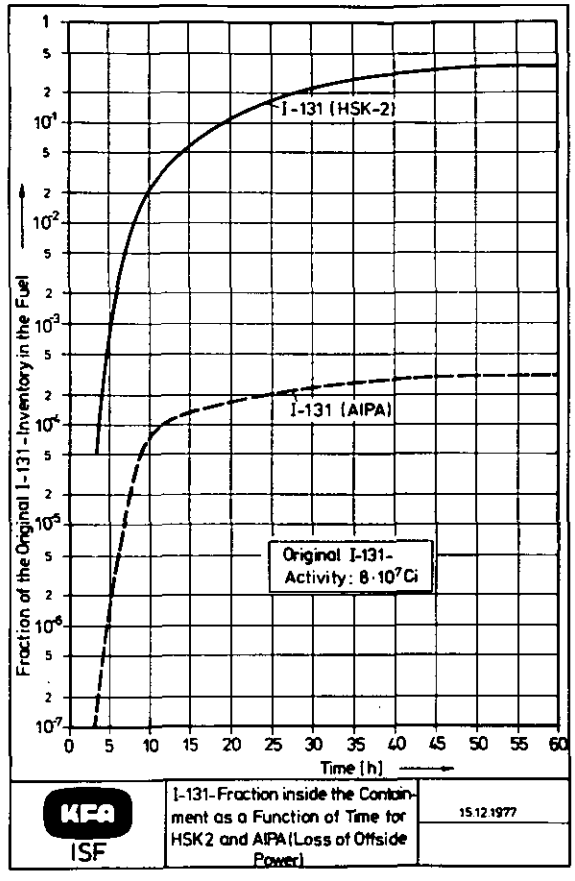


Fig. 12

The transport of the fission products into the containment is effected mainly by the slow expansion of the primary coolant due to the rising temperature. Convection contributes a varying part depending on the position of the leak.

As it is assumed that the liner cooling system also fails, the PCRV will heat up slowly, acting as a large heat sink. There will be a decomposition of concrete in the layers adjacent to the liner; however a failure of the PCRV can be excluded at least for the first few weeks. Therefore a failure of the liner would not have a significant influence on the course of the accident.

The containment does not fail as a consequence of the depressurization, as the design pressure and temperature are not exceeded. A failure is conceivable only as a long term event if there were no heat sink in the containment and the total after heat was eventually heating up the containment atmosphere causing an overpressure.

The fission products escape to the surroundings through the normal containment leakage. The radiation dose during the first 48 h in the vicinity of the reactor does not cause any immediate somatic effects (Fig. 13).

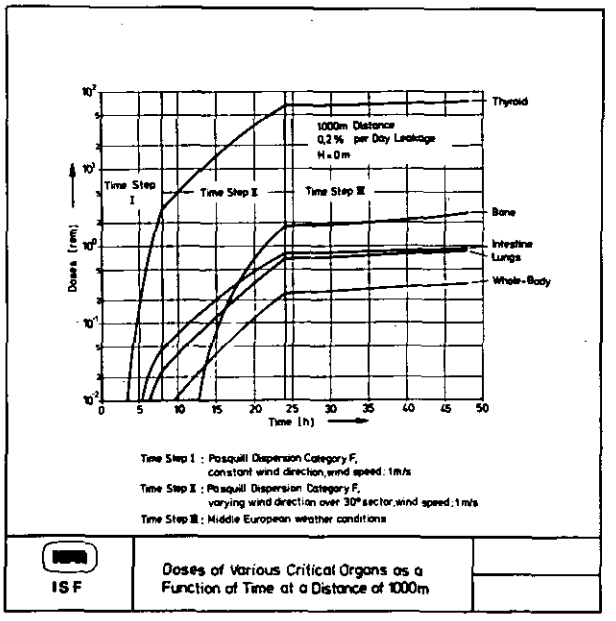


Fig. 13

Even if the containment would fail as a consequence of the accident, which would increase the dose by a factor of about 100, the consequences would not reach catastrophic dimensions.

Both the Loss of Offsite Power Accident and the hypothetical accident investigated by us finally lead to an unrestricted core heat up. A comparison of the consequences is therefore justified if one keeps in mind that a probabilistic study uses realistic data whereas our study is based on conservative values and assumptions. The consequences calculated in the AIPA Study do not differ considerably from our results. The main deviation is the much larger inhalation dose in our study caused by a very conservative plate-out factor.

5. Conclusions

The probabilistic risk analysis as well as the investigation of hypothetical accidents have demonstrated the good safety properties of the HTR. There can be no reasonable doubt that the high degree of safety is an inherent property of the system.

The main reasons are:

- the exclusive utilization of ceramic materials in the core and in the reflector warranting a high structural stability and the preservation of a coolable geometry;
- a simple cooling mechanism whose functioning continues even in the event of a rupture of the primary circuit or the reactor vessel;
- an equally simple after heat removal system whose functioning and efficiency can be tested under accident conditions;
- the high heat capacity of the reactor core which remains under accident conditions and prevents a destruction of the core by a fast temperature rise;
- the low energy stored in the primary circuit which cannot cause serious damage to the containment if the primary circuit breaks.

ECONOMIC EVALUATION OF THE HTR AS A POWER PLANT AND SOURCE OF PROCESS HEAT

by

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1. Introduction

The introduction of a new energy system, like the HTR will have to prove immidiate benefits or at least in the long run to promise superior economics.

The long term aspects are of particular significance as the time horizon will stretch for almost 40 years when planning and construction are added to the operating lifetime of the plant. For such a long period of time the development of the various cost factors weighs heavily and are correspondingly difficult to predict. It appears however, that increased importance will be attached to the areas of resource conservation and environmental protection. Costs related to these areas should therefore be viewed as particularly sensitive.

In this paper we have concentrated on the economics of the various fuel cycles, the impact of capital cost and the role of operating costs. Within this frame the HTR systems have been compared with a standard PWR as at present built in Germany. Some of the intangible economic areas like availability, siting problems, resource considerations are only mentioned in passing, although they may be of the greatest importance.

The re-evaluation of the German HTR-programme during the year 1977 also caused the establishment of a new cost basis.

The HTR construction companies performed a parallel evaluation of the plant costs of HTR direct cycle, HTR steam cycle and PWR. The KFA has jointly with the companies conducted an analysis of electricity production cost /1/. These findings are presented in this paper.

2. Methods

A thorough analysis will have to cover the construction period and the whole operating lifetime of the plant. As all cost items are likely to change due to inflation or other escalating factors these effects will have to be accounted for.

The methods for such dynamic calculations are based on the present worth principles and these have been deployed also for the joint study. A constant cost per kWh was defined over the whole lifetime. Although this present worth averaged cost is too high during the first years and too low in later years it provides a key figure for the comparison of overall economics of competing alternatives.

3. Cost Basis and Assumptions

The calculations were performed for German conditions with 1977 data and the results are given in DM or Dpf. Prices commonly quoted in US$, as for uranium ore and enrichment services, have been transferred at an exchange rate of 1 $ = 2.50 DM.

The plant costs are the single most important factor in the economic comparison. For German PWRs the capital costs are well established. For HTRs these have recently

been evaluated in a detailed study by the reactor companies. The study was based on a reactor with a thermal output of 3000 MW and alternatively with a gas turbine in a direct cycle or a steam turbine. In Table 1 a minor scaling of the plant size to yield the same electric output has been carried out to facilitate the comparison.

There are reasons to believe that an HTR with a direct cycle could be constructed cheaper than a LWR, but this has not yet been reflected in the cost assumptions. The plant costs in the table refer to a mature system without "first-of- a -kind" costs. No credit has been given for the dry cooling of the HTR1 system and neither has the potential to deliver some 400 MW heat for district heating been booked as an economic asset.

The costs of operation include personnel, maintenance, insurance, overheads and auxilliary items. These have been calculated in detail for a large PWR station. Although some variations exist in the various cost positions between the systems they have been assumed identical and to amount to 0.7 Dpf/kWh as of 1977.

The fuel cycle spans the services from uranium mining to reprocessing and waste disposal. The uranium ore has been assumed to cost 30 $/lb$U_3O_8$ per 1.1.1977 and to escalate either with the general inflation rate of 5 %/a or 7.5 %/a to reflect a real increase (2.4 %/a) in value due to a rising demand. The latter assumption is in line with recent Canadien and US estimates of the future uranium price. All general date common to the fuel cycle of HTR and PWR are listed in Tab. 2.

The total out-of-pile time of fuel from discharge to reloading of refabricated elements is at present estimated to 2 1/4 years. This figure is of particular significance to near-breeder systems and it is hoped that the turn-around-time can be reduced in the future.

The technology and costs of fabrication of fresh fuel are well known both for LWR and HTR (table 3). The additional cost penalty for refabrication of hot recycle fuel reflects the uncertainty in the estimates for an automized remote-handled production line.

Reprocessing and waste disposal are technologies for which commercial experience is still lacking. The data for LWR fuel is taken from a recent German evaluation /2,1/. For HTR fuel similiar data have been assessed on a corresponding cost basis, table 4. Fuel elements with higher heavy metal loadings and shorter burnup have lower specific processing costs. The estimates for long term retrievable storage of spent fuel are still uncertain but reflect todays best understanding.

The value of recycled material has been determined either from its specific reactor physics utilization in the same reactor or from credit calculation. In case a credit is given this is based on equivalence values for fissile plutonium and U 233 when used in thermal reactors.

4. Economics of Electricity Production

In accordance with the methods discussed above the whole reactor lifetime is included in the comparison and the fuel cycle and generating costs are present worth averages. Similiar inflation effects as assumed during the construction phase have been continued during the subsequent operating period. The start-up is set to 1.1.1988 and all present worths refer to that point in time.

The results are presented in three sections

- fuel cycle costs for once-through operation
- fuel cycle costs for recycle operation
- total electricity generating costs.

All HTR results apply to a dirct cycle plant with 41.4 % efficiency and the corresponding fuel cycle data for the steam cycle can be calculated by adding 8 %.

In the once-through operating mode the HTR is found to be more economic than the PWR. The fuel cycle costs (FCC) show a 20 - 25 % cost advantage with the largest benefit for the thorium/high enriched uranium cycle HO, /3/ Table 5. It is interesting to note that the differences stem to equal parts from the fuelling costs and the handling costs. Although the costs of handling HTR fuel in the fabrication and storage stages are higher per unit of heavy metal the resulting effect on electricity costs is less than for LWR fuel. In the HTR the burnup is higher by a factor of more than 3 and the thermal efficiency by almost 4/3 so that the electricity produced per kgHM is roughly four times higher than for the PWR.

The HTR fuel cycle with denatured uranium, MO, causes only a small cost penalty of 3-4 % compared to the HO and LO cycles.

The effect of varying the assumption for the future ore price from a 2.4 % real increase per year to a constant real level (i.e. 5 %/a nominal due to inflation) is small for once-through cycles as no credit for discharged fuel is taken into account and the gross uranium consumption for the two systems is not too dissimiliar. Apart from the estimates for the storage of spent fuel elements the costs appear to be supported by a sound basis and are believed to reflect with some accuracy the relative competitivness of HTR once-through fuel cycles versus present PWRs.

Reprocessing and recycle of bred fissionable material improve the economy of thermal reactors. The improvement in the FCC is higher for reactors with a large content of fissile isotopes in the discharged fuel elements as for instance, the high converting thorium cycles/3/. The savings in fuel, however, must been seen together with the increased costs in refabricating active material. In

Table 6 various HTR fuel cycles are compared with the PWR U/Pu system. The latter is based on a simulated recycle with credit assumed for Pu and U according to equivalence values and enrichment levels.

The HTR shows a cost advantage of 25 to 30 % for the HRS and HRM cycles. A core lay-out with a moderately high conversion ratio of some 0.75, like the HRM, commands superior economics over a wide range of assumptions. The high converting cycles of the PB/NB system are somewhat cheaper than the PWR for the real increase in the uranium ore price shown in the table whereas at a constant price the advantage is reserved.

A shorter out-of-pile time for recycled fuel is of particular importance to systems with high cycle inventories. The reduction from 2 1/4 year to 1 year decreases the FCC for the NB by 10 % but only by 2 % for the PWR.

An area of uncertainty is the cost assumptions for closing the fuel cycle for the various reacots. The head-end stage in the reprocessing of HTR fuel is thought to be of particular cost significance. Although due consideration was given to contingencies in the basic estimates it is interesting to note that even an additional 100 % in the head-end costs would increase the total contribution of reprocessing and disposal by only 30 % (Table 4). As a consequence the FCC for the HRM cycle, for instance, would rise by 6 % and not severly affect its competitive position. A similiar argument can be developed for the refabrication costs. Within all reasonable ranges of uncertainties the HTR offers a clear benefit with respect to fuel cycle economics.

The total generating costs for nuclear stations will always depend strongly on the capital costs. When comparing the total present worth of all costs of electricity

production over the wohle of the operatinglifetime, as is the underlying philosophy of the calculational methods used here, this effect is somewhat reduced compared to a calculation for the first year of operation only. This is due to inflationary increases in all variable costs positions during the plant lifetime. But still the capital costs make up half of the total.

In the once-through operating mode the HTR would generate electricity 4 to 8 % cheaper, depending on uranium escalation rate and direct or steam cycle, than present PWRs. The same cost advantage is found with recycle in HTR converters, whereas the generating costs of the PB/NB NB system would be equal to or slightly higher than the PWR with U + Pu recycle. The effect of variations in the plant costs is shown in Figs. 1-4. The HTR is competitive up to a cost penalty of 15 % on the plant costs.

The impact of future developments, such as a trend towards dry cooling and a stronger increase in the costs of energy resources, are likely to improve the economics of HTRs for electicity production even further.

5. The economics of Nuclear Coal Gasification

5.1 Introduction

The HTR system is able to supply heat at high temperature for utilization in chemical processes. Of the processes with the largest near-term realization prospects the gasification of hard coal and lignite have been studied in detail in Germany. Within the programme of the PNP project a market and economics evaluation has been carried out. The results of this study are discussed briefly below /4/.

5.2 Market potentials

In Germany 3/4 of the final energy is used for heating purposes and only 1/4 to provide light and power. At present the bulk of the heat market is dependant on petroleum products and in particular on light oils. Various recent forecasts arrive at the conclusion that oil will be in short supply before the turn of the century. The necessary substitution process is likely to take place in two stages, - firstly, natural gas will replace oil for heating in industry and private sectors, and - secondly, synthetic products based on coal, like "substitute natural gas" SNG, will gradually take over when natural gas resources are depleted.

The peak of gas production from natural resources is likely to be reached some two decades after oil production has culminated. The change-over to a new energy-system requires new technical equipment on the production side as well as at the consumer. The two stage substitution strategy is of great importance as during the "natural gas"-phase the changes take place on the consumer side and the investments are directed to expand the gas distribution network. When nuclear coal gasification enters the market the investments will be in plants on the production side.

The PNP study identified a potential market in Germany for SNG and the nuclear/chemical heat pipe system corresponding to 12 plants in the year 2000 and 52 plants in 2025. Of the total roughly 2/3 are for nuclear coal gasification and each plant is equipped with a 3000 MW(th) high temperature reactor. In the light of recent evaluations for the WEC 1977 of the ultimate exploitable resources of natural gas the figures appear too high for the initial period, although the long term necessity for substitute energy carriers is undisputed.

5.2 Economic Assumptions and Results

The cost basis for the economic evaluation is 1976 and all data and results are given in German monetary values of that year. The calculations were performed for two different sets of ground rules

- fictitious start-up of a large plant in 1976 and comparison with todays market
- start-up in year 2000 with escalation effects accounted for in the future market situation.

The costs of SNG production for the first year of operation as well as the present worth averaged costs over the 20 years lifetime have been compared for the various gasification processes.

The basic cost assumptions are given in Table 7 for two different gasification processes. Recent thinking favours a combination of hydro- and steam gasification for both brown coal and hard coal.

The results for a SNG production plant with a fictitious start of operations in 1976 are given in Table 8. The costs for the first operating year show that nuclear gasification is cheaper than conventional methods like the Lurgi process, but not competitive with natural gas and light heating oil at present prices.

The competitiveness of nuclear process heat should be judged against the economic situation during its large scale introduction phase, say, the year 2000.

All costs of coal gasification are assumed to escalate by 6 %/a. Due to the anticipated depletion of petroleum resourcesthe price of oil and natural gas is supposed to

be subject to an additional escalation of 2 %/a, i.e. 8 %/a in nominal terms.

There is a clear cost advantage for nuclear gasification at the turn of the century (Table 9). This is evident for brown coal already at the time of start-up whereas for hard coal the cost advantage will be realized over the lifetime.

6. Conclusions

The HTR power plant promises electricity production costs lower than present PWRs. The cost advantages lie primarily in the more favourable fuel cycle. The competitive economics are maintained over a wide range of cost assumptions.

In Germany there is a substantial market for nuclear process heat. The HTR can supply this market at competitive prices when the substition of petroleum products becomes necessary around the turn of the century.

The general characteristics of the HTR system with respect to anticipated future developments are likely to turn some of todays quantative and intangible arguments into strong economic incentives.

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MAIN DATE FOR LARGE POWER PLANTS AS OF MIDDLE 1977				
		HTR DIRECT CYCLE	HTR2 STEAM CYCLE	PWR
NET ELECTRIC OUTPUT	MW	1228	1228	1228
NET THERMAL EFFICIENCY	%	41.4	38.3	32.6
COOLING TOWER		DRY	WET	WET
PLANT COST	MIO DM	1930	1900	1870
OWNERS COSTS	MIO DM	180	180	180
CONSTRUCTION TIME	A	6.5	6.5	6
DECOMMISSIONING COSTS	MIO DM	300	300	170

TABLE 1

FUEL CYCLE COST ASSUMPTIONS PER 1.1.1977	
URANIUM ORE	30 $/LBU_3O_8$
THORIUM	20 $/KG TH
ENRICHMENT	110 $/SWU
U 233 EQUIVALENCE	1.25 U(93 %)
PU-FISS EQUIVALENCE	0.60 U(93 %)
ESCALATION	5 % P.A.
INTEREST & DISCOUNT RATE	8 % P.A.
TAX ON FISSILE	2 % P.A.
OUT-OF-PILE TIME	2,25 YEARS

TABLE 2

FUEL FABRICATION COSTS AS OF 1.1.1977	
LWR:	
FRESH FUEL	340 DM/KG HM
HTR:	
FRESH LEU C.P.	365 DM/KG HM
FRESH TH/U MOX C.P.	355 DM/KG HM
FRESH HEU FEED C.P.	2550 DM/KG HM
BALL PRESSING INCL. GRAPHITE	5.80 DM/FE
DUMMY BALL	3.54 DM/FE
RECYCLE PENALTY C.P.	+ 300 %
RECYCLE PENALTY BALL	+ 100 %

TABLE 3

COST OF REPROCESSING AND SPENT FUEL STORAGE (DM/KG HM) AS OF 1.1.1977		
	LWR	HTR (1
SHIPPING	40	175
HEAD-END	190	740
SOLVENT EXTRACTION PRODUCT CONVERSION	280	600
OFF-GAS AND WASTE TREATMENT	300	1070
WAST SHIPPING AND STORAGE	90	
TOTAL REPROCESSING	900	2585
SPENT FUEL STORAGE	1060	2300
1) BASED ON THTR FUEL WITH 11 G HM/BALL AND 100 000 MWD/T		

TABLE 4

ONCE-THROUGH FCC /DPF/KWH/, PRESENT WORTH AVERAGE 1988 INFLATION RATE 5%/A, URANIUM ORE 7.5 %/A				
	PWR	LO	MO	HO
FUEL WITH FINANCING AND TAX	4.01	3.27	3.19	3.27
FABRICATION	0,42	0.31	0.35	0.29
SPENT FUEL STORAGE	1.18	0.74	0.92	0.68
TOTAL FCC	5.61	4.32	4.46	4.24

TABLE 5

CLOSED CYCLE FCC /DPF/KWH/, PRESENT WORTH AVERAGE 1988 INFLATION RATE 5 %/A, URANIUM ORE 7.5 %/A					
	PWR(1	HRS	HRM	PB	NB
FUEL INCL. CREDIT	2.92	2.25	1,94	1,65	1,63
FABRICATION	0.42	0,41	0,49	0,95	1,04
REPROCESSING AND DISPOSAL	1,00	0,68	0,63	1,54	1,25
TOTAL FCC	4.34	3,34	3,06	4,14	3,92
1) SIMULATED RECYCLE WITH CREDIT FOR DISCHARGED FISSILE MATERIAL					

TABLE 6

FUEL	CYCLE	TYPE	PRESENT WORTH LIFE FCC (DPF/KWH) ORE PRICE	CONST	ORE PRICE	+ 2,4 %/A
O	U/U	PWR	4,66	100 %	5,61	100 %
P	TH/U 93 %	HO	3,62	78 %	4,24	76 %
E	TH/U 20 %	MO	3,81	82 %	4,46	79 %
N	U/U	LO	3,62	78 %	4,32	77 %
C	U/U + PU	PWR	3,74	100 %	4,34	100 %
L	TH/U 93 %	HRS	2,96	79 %	3,34	77 %
O	TH/U 93 %	HRM	2,85	76 %	3,06	71 %
S	TH/U 93 %	PB/NB	3,95	105 %	3,95	91 %
E D	TH/U 20 %	MR	3,03	81 %	3,46	80 %

TABLE 6 A

BASIS ASSUMPTIONS FOR COST CALCULATION, 1976 DATA				
			HKV[1] BROWN COAL	WKV[1] HARD COAL
THERMAL REACTOR POWER		MW	3000	3000
ANNUAL UTILIZATION		H/A	7500	7500
COAL INPUT	10^6	GCAL/A	34.76	22.22
GAS OUTPUT	10^6	GCAL/A	28.22	22.24
REST COKE	10^6	GCAL/A	14.20	-
NET ELECTRICITY PRODUCTION	10^6	KWH/A	877	6120
PLANT COSTS	10^6	DM	3615	3288
COAL PRICE		DM/GCAL	7.00	20.63

[1] HKV: HYDROGASIFICATION, WKV: STEAMGASIFICATION

TABLE 7

SNG PRODUCTION COST FOR 3000 MW PLANT WITH FICTITIOUS START-UP 1976	
	COSTS IN 1. YEAR OF OPERATION DM/GCAL
NUCLEAR GASIFICATION	
- BROWN COAL	29 - 32
- HARD COAL	40 - 45
CONVENTIONAL GASIFICATION	
- BROWN COAL	37
- HARD COAL	62
NATURAL GAS (FREE BORDER)	22
HEATING OIL (FREE REFINERY)	25

TABLE 8

SNG PRODUCTION COSTS FOR 3000 MW PLANT WITH START-UP 2000. DATA GIVEN IN 1976 MONETARY VALUE		
	1. YEAR OF OPERATION DM/GCAL	LIFETIME AVERAGE DM/GCAL
NUCLEAR GASIFICATION		
- BROWN COAL	29 - 32	37 - 39
- HARD COAL	40 - 45	51 - 60
CONVENTIONAL GASI-FICATION		
- BRWON COAL	37	46
- HARD COAL	62	88
NATURAL GAS (FREE BORDER)	34	65
HEATING OIL (FREE BORDER)	39	74

TABLE 9

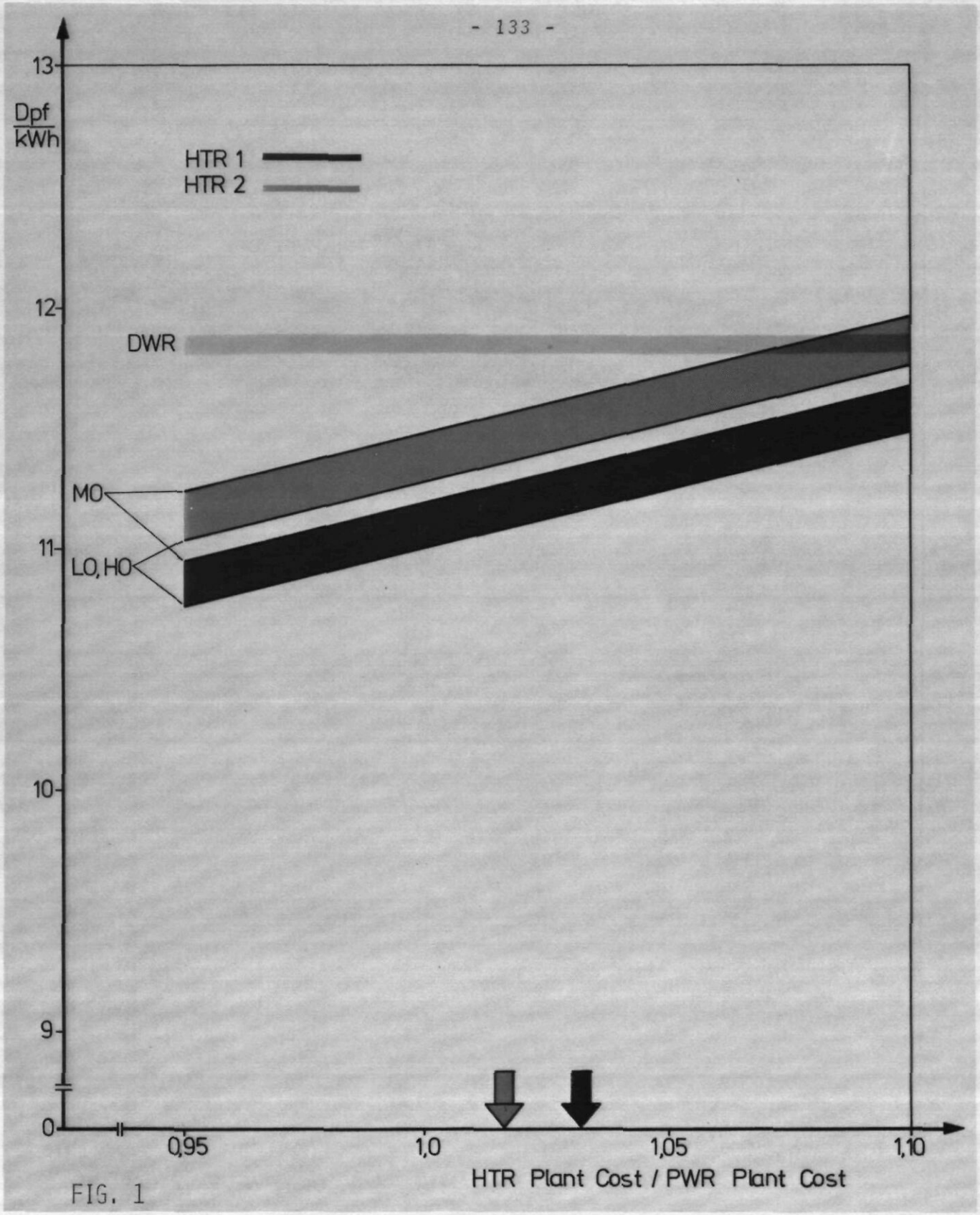


FIG. 1

Costs of Electricity Production

Once-Through Cycles with Start-up 1988

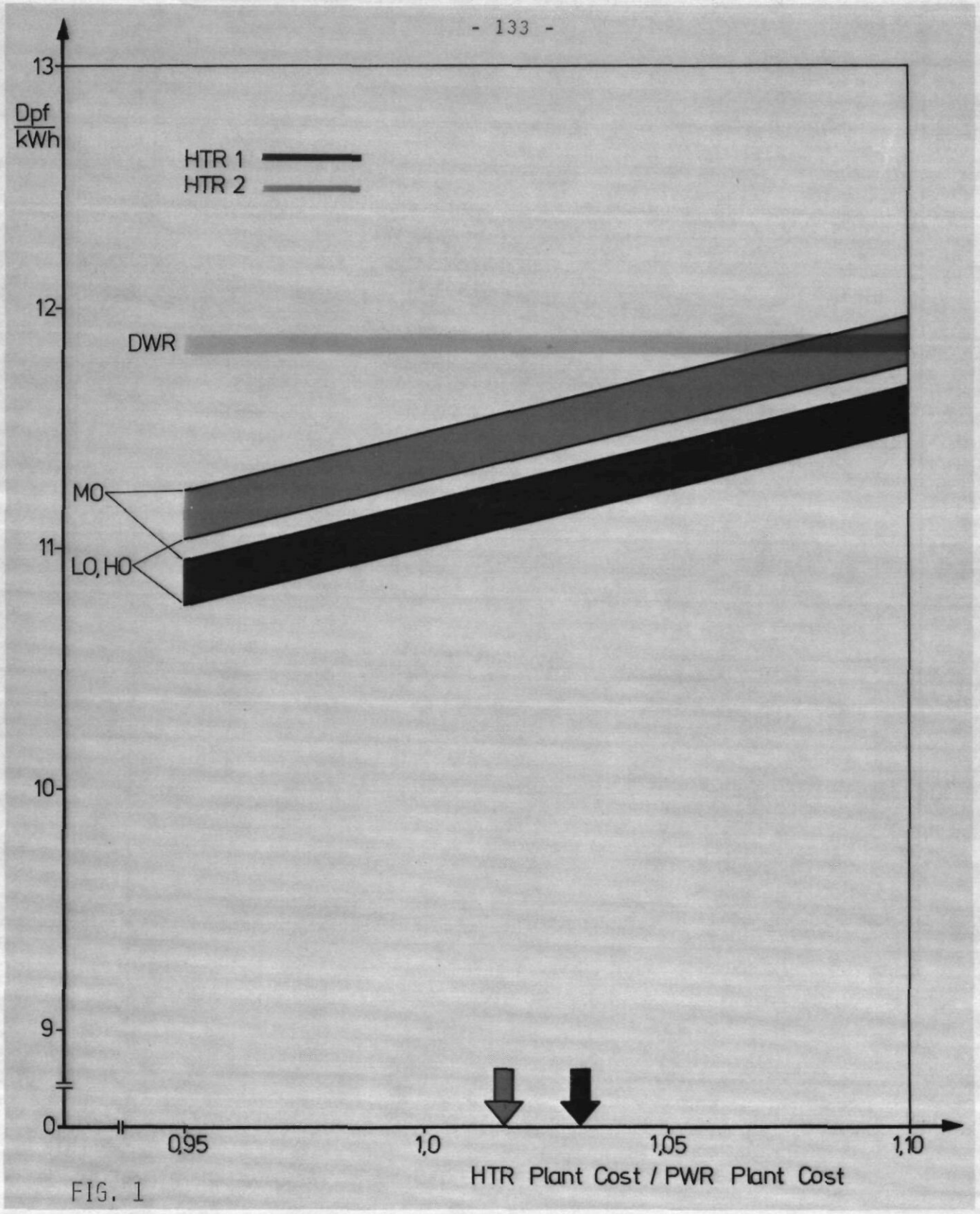


FIG. 1

Costs of Electricity Production

Once-Through Cycles with Start-up 1988

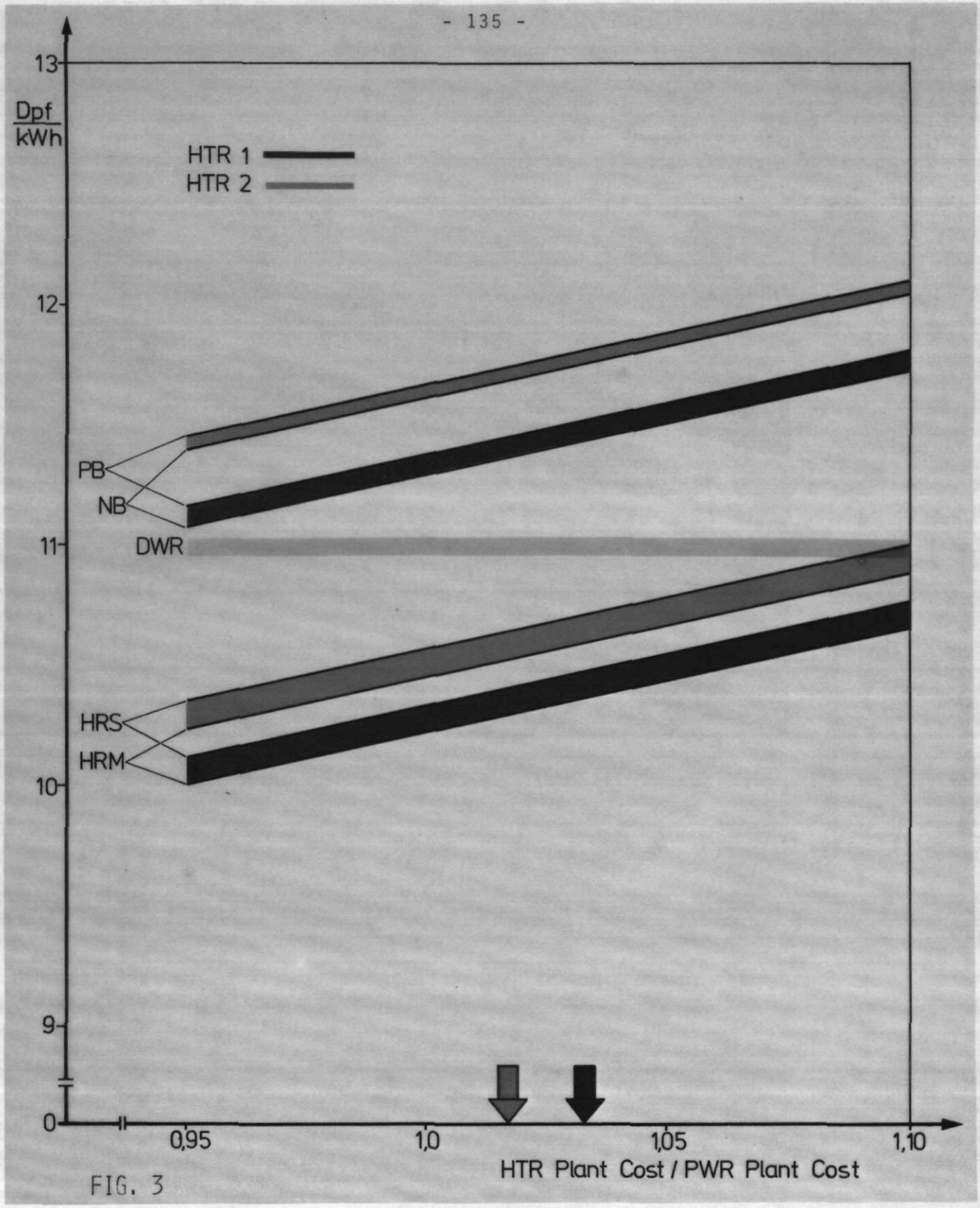


FIG. 3

Costs of Electricity Production

Closed Fuel Cycles with Start-up 1988

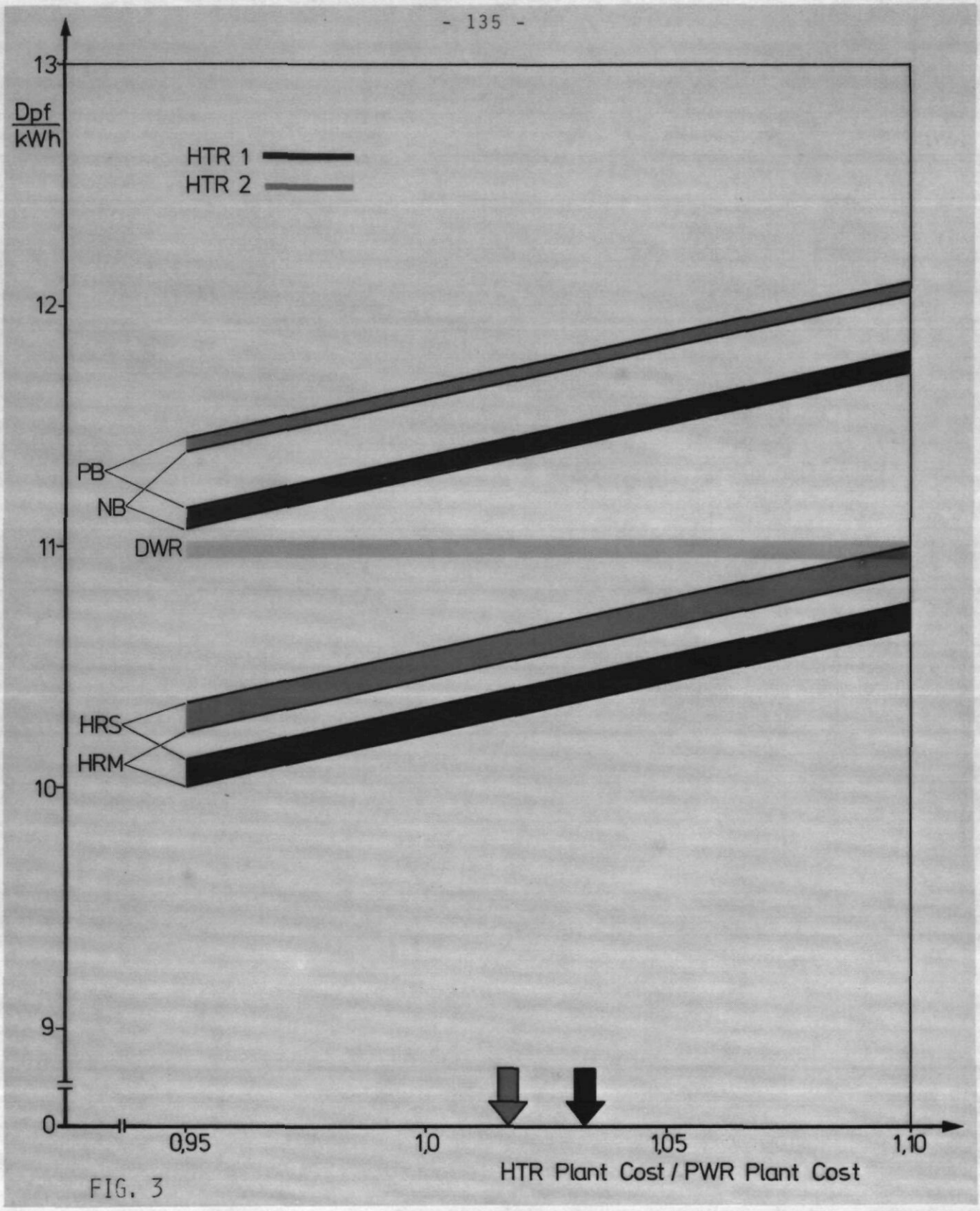


FIG. 3

Costs of Electricity Production

Closed Fuel Cycles with Start-up 1988

Dr. E. Balthesen
KFA - PTH

HTR-Introduction Strategy in FRG

The HTR development in the Federal Republic of Germany is based on the particular interest in nuclear process heat which in a first step can be utilized for the production of synthetic natural gas from coal.

In the development of the process-heat HTR, the electricity producing HTR presents itself as a technically less complicated intermediate stage which should be realized prior to the process-heat HTR, if possible.

Our prime motive for the further development of HTR lies in the direct generation and utilization of process heat at high temperature levels, which is possible only with this reactor system; the HTR offers the option of a major introduction of nuclear energy into the heat market.

Among various possible process heat applications, our main interest lies in the gasification of lignite and hard coal, to obtain synthetic natural gas. Coal is the only fossil energy carrier which can reliably be expected to be available in the Federal Republic of Germany in the long run due to our own considerable domestic resources and due to the huge world-wide deposits (several times the resources of mineral oil and natural gas). Coal will therefore have to remain a major contributor to our energy supply for a long time to come. It will however become increasingly necessary to meet the growing demand for "comfortable" and low-pollution end products by means of the appropriate conversion technologies. In this context the generation of fuel-gas and synthesis gas is particulary interesting.
Our strong interest in the process-heat HTR for coal gasification encounters the difficulty that the technological problems to be solved in the development of this particular HTR type are considerable and will therefore, require a corresponding amount of time and money. This can be underlined as two technologies must be developed to the prototype stage simultaneously: i. e. the reactor and the gasification plant. For these reasons it hardly seems possible that construction of a prototype plant can be started before the second half of the 1980s.

Quite apart from the process heat aspect, there are a number of arguments in favour of continued HTR development. None of these reasons, taken separately, is as striking as for example the fuel-stretching effect of the fast breeder reactor or the process-heat applications of the HTR; but taken together they provide a clear additional incentive for the development of the HTR system, including its applications for electricity production:

- The high coolant outlet temperature of the HTR gives high efficiency and thus a lower waste heat discharge to the environment and greater siting flexibility on account of lower cooling water requirements.

- In the case of direct cycle HTR with helium turbine (HHT)
  * the efficiency can be increased even more and
  * dry cooling can be introduced without reducing efficiency
  * the waste heat can be used for district heating without reduction of electric efficiency.

- In the closed uranium/thorium fuel cycle the HTR uranium requirements are only 60 % of those of the LWR with Pu recycle; in advanced HTR versions (high conversion) uranium consumption can potentially be reduced even further to approximately 10 to 15 % of present LWR uranium requirements.

- The HTR fuel cycle is extremely flexible and can be adapted continuously to changing conditions, even in an individual HTR power plant.

These reasons, which also (and some of them only) apply to the electricity producing HTR, taken together with the fact that the technical problems and the time required for the development of a first nuclear coal gasification plant are greater, have led us to consider the implementation of the HTR for electricity production as an intermediate stage preceding the construction of the first nuclear process heat facility. This approach should broaden the HTR's market basis. Consequently, the construction of a larger electricity producing plant is being planned for the first half of the 1980s.

All these incentives and the corresponding strategy require a simultaneous development for both applications, electricity and process heat. A program of of such an extent can only be performed within the frame of a national program, embedded as far as possible in international efforts. National program means that the main burden of the development in the introduction phase has to be carried by the public, by the government. Major contributions from the users can only be expected and demanded as far as a commercial utilization of the technology can be envisaged. As the public funding resources are limited today and in the future, a strong technical and organizational concentration is indispensable.

A technical concentration will be possible by the development and application of the same basic design concept for the nuclear heat supply system for both lines. This systems should have as many common features as possible for electricity production and for process heat application. In extension of the AVR-THTR-line and utilizing the specific temperature potential of the spherical fuel elements, this design will be based on the pebble bed core in a PCRV. Numerous R + D- tasks are common for both applications.

An organizational concentration will take place by a well defined task distribution between the participating industry groups (Fig. 1). BBC/HRB will take the lead for the power plant, the coal industry and the KWU-group will take the lead for the process heat plant and HRB and GHT will establish a consortium for the development of the nuclear heat supply system in order to ensure that the common trunk will be maintained and pursued. KFA will play a lead role in the R + D-area. The technical experts will be able to use their experience for both applications and it will be possible to maintain the technical capacity during the long introduction phase.

If we take into account the limited funding resources and development capacities and the fact that the confidence in this new technology has to be enlarged or partially to be established, the following introduction strategy is consistent and should be realizable (Fig. 2)

- Continous operation of the research reactor AVR at high gas outlet temperature (950^{o} C) represents a unique demonstration of the feasibility of the technology. The reactor, simultaneously a test facility for improved fuel elements, should be operated at least until the successful operation of THTR for a certain time period.
- Construction of the THTR-300 prototype power plant at Schmehausen will be continued with priority, in accordance with this project's significance in the context of the technical reorientation of the German HTR-program.

In the restructured HTR programme, THTR-300 holds a key position. Its construction has top priority among the German HTR activities. Thus, the industry involved has temporarily postponed specific activities for the future-oriented R + D programs. In the future, KFA-Jülich will also be increasingly involved in the work for this project.
The completion of the construction will certainly govern the time-schedules for the advanced projects.

- Therefore, construction of the next electricity producing plant cannot start before about 1983 as the prerunning THTR-300 must have first of all a sufficiently long period of successful operation.

Provided that the German utilities will follow the recommendations of the reactor industry and support the development, this next plant will be a 600 MW_{el} gas turbine plant. Following the steam cycle THTR-plant this design represents a reasonable scale up step and aims at a better utilization of the potential of the HTR. At the time being the technical features of a steam cycle plant are not sufficient attractive compared to the well established LWR, at least for the rather small German market. In order to reduce the risks of this advanced technology the support of the utilities will strongly depend on the proof of the feasibility of critical components. Therefore the successful operation of the 50 MW_{el} gas turbine plant EVO, the successful start of operation of the gas turbine and hot circuit test facility HHV, both in 1978, the proof of the feasibility of the warm liner and the inspectability of the gas turbine (decontamination) and the licensability of the whole system are major milestones in the planning phase.

- Construction of the process heat facility must not overlap too much with that of the electricity producing plant since neither the manufacturing industry's staff nor the government funds would be sufficient for fully parallel construction. However, the construction should be delayed as little as possible beyond attainment of the technical feasibility.

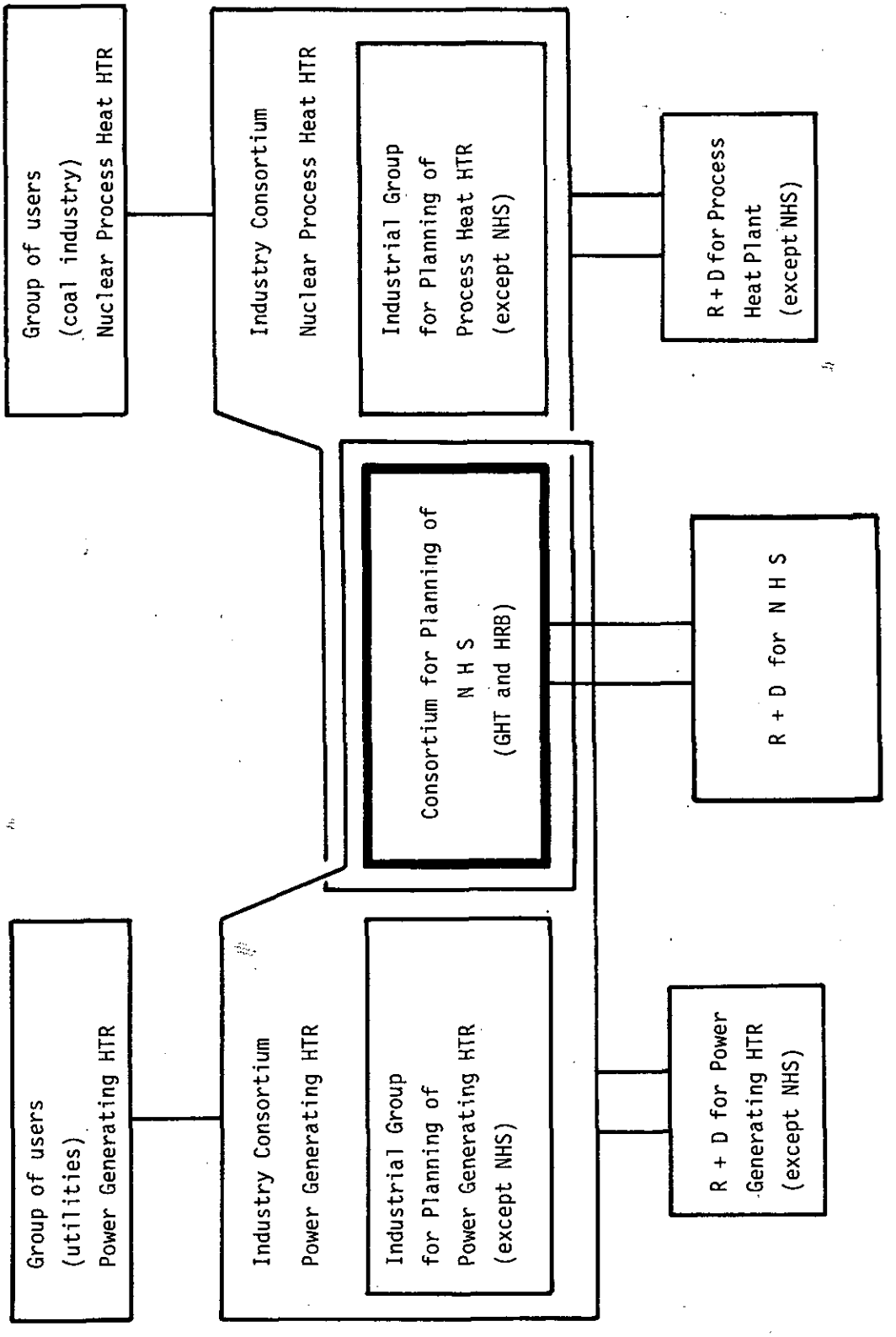
After conclusion of a conceptual phase the PNP project partners (coal industry, reactor industry, KFA) have decided to start a reference phase for the planning of a 500 MW_{th} prototype plant in 1978. This plant should have two loops with 250 MW_{th} each one for steam gasification of hard coal the other for hydrogasification of lignite. Predominent issues of the development program are the proof of the licensability of the combined reactor-gasification plant system and the successful operation of the semi-technical and prototype gasification plants. Start of construction has been scheduled for 1985.

Up to now the continuation of the HTR-introduction is only based on assumptions. Due to careful investigations and design studies it can be taken for granted that the construction of large commercial-size plants for both applications (3000 MW_{th}) can be performed on the basis of the construction and operation experience with the above mentioned demo- and prototype plants. In a first approach it has been assumed that experience from a part of the construction of the 600 MW_{el}-demo-plant, particularly the main licensing steps, are sufficient to justify the start of construction of a large power plant whereas, due to the more sophisticated technology, experience from the operation of the process heat prototype plant should be available before construction of a large NPH plant. Number, type and size of the commercial HTR's will be governed by the conditions of the energy market at the end of this century which cannot be foreseen in detail today. The main goal of the first phase is to demonstrate that the HTR technology is available and can be economically applied.

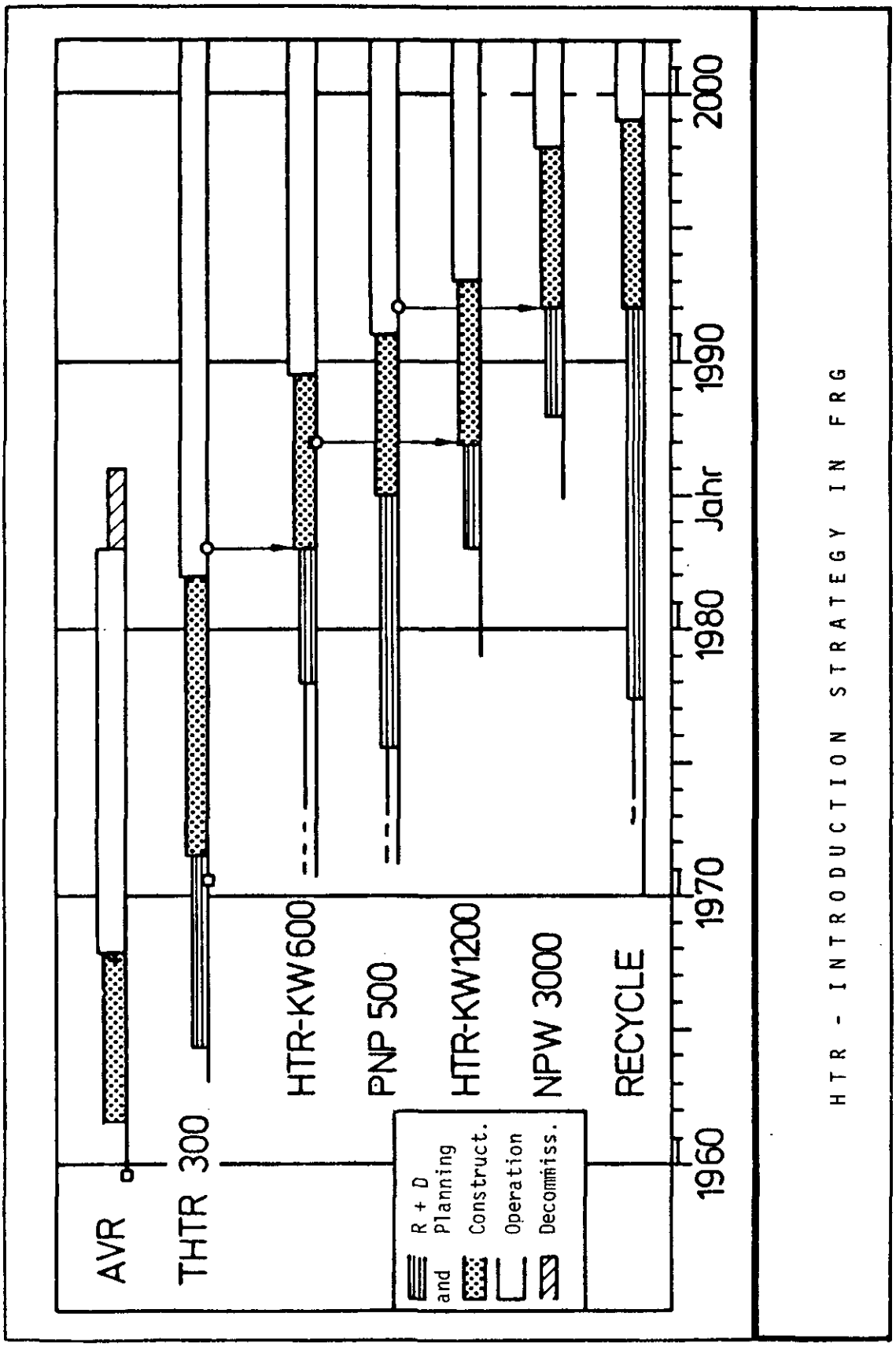
In view of the high cost of the development of a new reactor system, including fuel cycle and infrastructure, the German HTR development will benefit by an intense international cooperation.

The financial burden implied by the complete introduction of a new reactor system, including the necessary infrastructure, cannot be stated precisely; it could, if the establishment of the complete fuel cycle is included, lie within a 5 to 10 billion DM range. Projects of this scope require close international cooperation and a sharing of efforts with other countries. With this in mind, we have begun to intensify the existing basis of cooperation in the HTR field.

- In an extension of the already existing close industrial links with the United States, BMFT and US-ERDA signed an " umbrella " agreement on cooperation in the field of gas-cooled reactors in February 1977. At present, this agreement is progressively being implemented by specific cooperation arrangements. In September France and Switzerland also joined this agreement.
- On the basis of the joint statement made in February 1976 by the responsible ministers of the Federal Republic and of France concerning cooperation in the field of advanced reactor systems, the governments of both countries agreed on guidelines for this cooperation in May 1976, both for the fast breeder and for the HTR. In the meantime, concrete cooperation possibilities have been explored for the HTR. An exchange of technical details on the respective status of French and German activities has been initiated. This exchange is, above all, intended to ensure that the concept definition process under way in the Federal Republic can closely be followed by the French side.
- Cooperation with Switzerland has been going on smoothly and successfully since 1973 in the joint HHT project. Both sides wish to continue cooperation with regard to the HTR for electricity production.
- Finally, already existing links with Japan and Austria have recently been resumed and intensified. Contractual arrangements initially aiming at cooperation in specialized fields are being negotiated.


Group of users
(coal industry)
Nuclear Process Heat HTR
Industry Consortium
Nuclear Process Heat HTR
Industrial Group
for Planning of
Process Heat HTR
(except NHS)
R + D for Process
Heat Plant
(except NHS)
Consortium for Planning of
N H S
(GHT and HRB)
R + D for N H S
Group of users
(utilities)
Power Generating HTR
Industry Consortium
Power Generating HTR
Industrial Group
for Planning of
Power Generating HTR
(except NHS)
R + D for Power
Generating HTR
(except NHS)

(Fig. 1)


HTR - INTRODUCTION STRATEGY IN FRG
1960
1970
1980
1990
2000
Jahr
AVR
THTR 300
HTR-KW600
PNP 500
HTR-KW1200
NPW 3000
RECYCLE
R + D and Planning
Construct.
Operation
Decommiss.

(Fig. 2)